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AFITBUNKI: A MODIFIED ITERATIVE CODE
TO UNFOLD NEUTRON SPECTRA FROM
BONNER SPHERE DETECTOR DATA

THESIS

Sean C. Miller, Captain, USAF
AFIT/GNE/ENP/93M-5

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**AFITBUNKI: A MODIFIED ITERATIVE CODE TO UNFOLD NEUTRON
SPECTRA FROM BONNER SPHERE DETECTOR DATA**

THESIS

**Presented to the Faculty of the School of Engineering
of the Air Force Institute of Technology
Air University**

In Partial Fulfillment of the

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Requirements for the Degree of

Master of Science

Sean C. Miller, B.S.

Captain, USAF

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Preface

The purpose in modifying the BUNKI/BUNKIUT code was to develop a code that accurately unfolds Bonner sphere detector data so that neutron spectra can be characterized and dosimetric quantities can be measured in the lab in real-time. BUNKI was developed in 1983 with many user-selectable choices of response matrices and several unfolding algorithms. Since then, BUNKI and BUNKIUT have been used to evaluate the accuracy of these choices. In AFITBUNKI, BUNKI's most effective features have been 'hardwired' in and several features have been added such as incorporating a selection of initial-guess spectra and a number of the new dosimetric conversion factors which have been defined since 1983.

In developing the AFITBUNKI code and writing this thesis, I have been given a great deal of help from others. I am very grateful to my advisor, Lt Col Richard S. Hartley, for his guidance and support throughout this project and degree program. I would like to thank Dr. Nolan E. Hertel for providing data, insight to theory, and feedback on my work; without him, much of the struggling would be ongoing. I wish to thank Dr. George John who taught me the fundamentals of Nuclear Physics and Dr. Kirk Mathews who taught me the fundamentals of Nuclear Engineering. A word of appreciation is owed Capt Charlie Brennan for his help with portions of the MCNP modeling and to Mr. Bob Hendricks for his extensive assistance in the lab. Finally, I wish to thank all of my classmates who made the AFIT experience more than just a good time.

Sean C. Miller

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Abstract

The neutron spectrum unfolding code BUNKI, developed at the Naval Research Laboratory in 1983, was modified to incorporate a finer energy group structure, seven initial-guess spectra, and new dose conversion factors. The modified code, AFITBUNKI, unfolds spectra into 54 energy groups between 10^{-11} and 14.92 MeV similar to those identified in ISO standard 8529. The code calculates fluence, absorbed dose, percent of effective dose equivalent, and percent of ambient dose equivalent as a function of neutron energy. In addition to these spectral quantities, AFITBUNKI calculates total energy-integrated fluence (Φ), absorbed dose (D), dose equivalent (H), effective quality factor (\overline{Q}), ambient dose (D^*), both *ICRP Publication 26*- and *ICRP Publication 60*-based ambient dose equivalent (H^*) and effective ambient quality factor (\overline{Q}^*), effective dose equivalent (H_E), effective dose (E), and the average neutron energy (\overline{E}). AFITBUNKI uses BUNKI's SPUNIT iterative unfolding algorithm and UTA54, the 171-energy group response matrix developed at The University of Texas at Austin using ENDF/B-IV cross sections collapsed into the 54-group structure. The user specifies an initial spectrum or directs the MAXIET algorithm to calculate a $(1/E)$ and Maxwellian spectrum as an initial guess. The modified code is verified against spectra unfolded by BUNKIUT, the personal-computer version of BUNKI, and validated against calculations of a NIST fit of ^{252}Cf fission neutrons and against calculations of both D_2O - and polyethylene-moderated ^{252}Cf fission neutrons made by Los Alamos National Laboratory's Monte Carlo Neutron-Photon transport code, MCNP.

AFITBUNKI: A MODIFIED ITERATIVE CODE TO UNFOLD NEUTRON SPECTRA FROM BONNER SPHERE DATA

1. Introduction

The neutron spectrum unfolding code BUNKI [21:6] has been modified to create a more current version called AFITBUNKI. AFITBUNKI is primarily intended to unfold neutron spectra to calculate dosimetric quantities. The objective in modifying BUNKI was to implement the most appropriate unfolding algorithm and response matrix, restrict the energy range to one most applicable to many health physicists, refine the spectral resolution by increasing the number of energy groups, add a selection of initial-guess spectra for use in the iterative unfolding process, and add new dosimetric conversion factors. It was also thought that by recollapsing the response matrix, more accurate thermal neutron measurements could be made. Developed on a SUN SPARCstation and written in FORTRAN 77, AFITBUNKI is a reliable neutron spectrum unfolding code which is readily transported to a personal computer. Appendix A contains a listing of the AFITBUNKI code.

Applications of AFITBUNKI

In 1960, R.L. Bramblett, R.I. Ewing, and T.W. Bonner described a neutron spectrometer which consisted of thermal neutron detectors covered with polyethylene spheres of various diameters [1:1]. In principle, as neutrons traverse the sphere, epithermal and fast neutrons scatter off the polyethylene losing energy until they either reach thermal equilibrium or leave the moderator. The detector will respond to thermal neutrons which arrive at its location at the center of the moderating sphere. Each detector-moderator combination will have a different response to neutrons as a function of energy. A spectrum from sphere count rates can be unfolded providing information about the energy distribution of the incident neutrons [24:516]. This type of neutron spectrometer is commonly called a Bonner Sphere Spectrometer and the polyethylene spheres are called Bonner spheres.

AFITBUNKI is a neutron fluence spectrum unfolding code which unfolds Bonner sphere data by employing the user's *a priori* knowledge and a Bonner sphere detector response matrix to solve a numerical approximation of the Fredholm integral equation of the first kind. Its primary intended user is health and research physicists performing dosimetric evaluations.

AFITBUNKI, like BUNKI [21:6], calculates fluence as a function of energy via the unfolding process. In accordance with recommendations set forth by the International Commission on Radiation Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU), it also calculates total fluence (Φ), absorbed dose (D), dose equivalent (H), average quality factor (\bar{Q}) and average neutron energy (\bar{E}). In addition AFITBUNKI goes further than BUNKI and BUNKIUT and calculates percent

of effective dose equivalent, and percent of ambient dose equivalent as a function of neutron energy, and calculates total energy-integrated ambient dose (D^*), both *ICRP Publication 26*- and *Publication 60*-defined ambient dose equivalent (H^*) and corresponding effective ambient quality factors (Q^*), as well as effective dose equivalent (H_E), and effective dose (E).

Objective and Approach

AFITBUNKI is a modification of the neutron spectrum unfolding code BUNKI, developed at the Naval Research Laboratory in 1983. In 1986, BUNKIUT, the personal-computer version of BUNKI, was developed at The University of Texas at Austin [10:--]. It incorporated the same features as the original BUNKI, as well as an optional plotting routine. The objective of the current work was to modify BUNKIUT to better meet the current needs of health and research physicists. This was accomplished by retaining the best features of BUNKI and BUNKIUT, eliminating those which have been shown to be less effective, and incorporating new dosimetric conversion factors.

The energy range in AFITBUNKI extends from 10^{-11} to 14.92 MeV, a range suitable for most health physics applications. The Bonner sphere response matrix, also developed at the University of Texas at Austin using ENDF B-IV cross-section data [9:509], has been incorporated because it is the most current response matrix available for $^6\text{LiI(Eu)}$ Bonner sphere detectors. Spectral resolution in AFITBUNKI has been refined by binning the energy interval into twice as many energy groups as those used in BUNKI and BUNKIUT. Additionally, by extending the thermal group down to 10^{-11} MeV from 10^{-8} MeV, more accurate thermal responses are achieved.

AFITBUNKI retains the iterative unfolding algorithm, SPUNIT [2:--], and an algorithm which calculates a Maxwellian plus a 1/E initial guess spectrum, MAXIET [21:4]. New features introduced in AFITBUNKI include a selection of seven user-specified initial guess spectra for use in the unfolding process. A final objective in modifying BUNKIUT was to be able to calculate the most recent dosimetric quantities by including the the most recent fluence to dose conversion factors defined by the ICRP and ICRU.

Starting with BUNKIUT, AFITBUNKI was developed in several steps. The first was to identify BUNKIUT's constituent routines and algorithms and select those which were to be retained. A 54 energy-group binning structure which corresponds closely to one recommended in International Standardization Organization (ISO) Standard 8529 [19:--] was specified and the 171 energy-group response matrix was then collapsed into the new energy binning structure. Seven neutron fluence spectra were identified for incorporation as initial guess spectra into AFITBUNKI. Dosimetric conversion factors were interpolated to fit the new binning structure as well. AFITBUNKI was then coded in FORTRAN 77 with a more user-friendly front-end.

AFITBUNKI was verified by unfolding three well-characterized reference spectra and comparing them to spectra unfolded by BUNKIUT. AFITBUNKI was validated by comparing the same three unfolded spectra to spectra calculated using either an analytical fit or a neutron transport code. The three reference spectra included the The National Institute of Standards and Technology (NIST) analytical fit of a bare ^{252}Cf fission spectrum, and fission spectra of both D_2O - and polyethylene-moderated ^{252}Cf calculated using Los Alamos National Laboratory's Monte Carlo Neutron-Photon transport code, MCNP. The quantities compared in verifying and validating AFITBUNKI were average

energy, \overline{E} , and energy-integrated values of fluence, Φ , absorbed dose-to-fluence ratio, H/Φ , and ambient dose-to-fluence ratio, H^*/Φ .

II. BUNKI History

BUNKI and the personal computer version, BUNKIUT, are a widely accepted neutron spectrum unfolding codes which have been used by researchers for almost a decade. SPUNIT, one of BUNKI's two iterative unfolding routines was the one selected for use in AFITBUNKI. A brief description of BUNKI and its history provides a starting point for developing AFITBUNKI.

Description of BUNKI

In 1983, K.A. Lowry and T.L. Johnson of the Naval Research Laboratory developed the BUNKI neutron unfolding computer code written in FORTRAN IV for use on a DEC-10 computer [21:6]. BUNKI, designed for laboratory use, unfolds a neutron fluence spectrum and calculates absorbed dose, and dose equivalent spectra as a function of neutron energy. It calculates total energy-integrated fluence, absorbed dose and dose equivalent as well as the quality factor, and average neutron energy. The user of the code chooses to unfold the spectrum using one of two methods, either the "SPUNIT" or "BON31G" algorithm. SPUNIT and BON31G were both written as stand-alone unfolding codes which use iterative recursion methods to unfold the spectrum by minimizing the deviation between measured and calculated detector responses [21:2]. Their incorporation into BUNKI allows the application of any *a priori* knowledge of the spectrum such as the spectral shape. The user can specify an initial spectrum or direct a

subroutine, MAXIET, to calculate a Maxwellian and 1/E spectrum as an initial guess. BUNKI allows the user to select one of nine response matrices to unfold spectra [21:4].

The SPUNIT unfolding routine was selected as the only one to be used in AFITBUNKI because it has been shown in studies completed by Johnson to converge to a solution approximately 20% faster than BON31G [20:3].

To enable the user to have a selection of response matrices, all of the matrices used in BUNKI have been binned into the same energy intervals. T.L. Johnson *et al.* chose the same 31 energy groups between 10^{-8} and 400 MeV used in the SAN4 response matrix. SAN4, constructed by R.S. Sanna in 1973, describes responses to a 4 mm x 4 mm LiF detector [25:1]. In order to force the other response matrices used in BUNKI into the same energy binning structure, Johnson collapsed, re-binned, and sometimes spliced together response matrices [23:3-5]. One such response matrix was UTA4, also calculated for a 4 mm x 4 mm LiF detector. UTA4 is based on a response matrix calculated by N.E. Hertel and J.W. Davidson at The University of Texas at Austin, which was originally 171 groups and extended from 10^{-11} to 17.3 MeV [9:509]. For its incorporation into BUNKI, Hertel and Davidson's matrix was collapsed into the first 26 (10^{-8} to 17.3 MeV) of the 31 SAN4 energy groups. To complete the matrix groups 27 to 31, (17.3 to 400 MeV), SAN4 response matrix values were appended by Johnson [23:6].

In 1983 Johnson *et al.* completed a study to evaluate the effect of the choice of response matrix on the results of the unfolded Bonner sphere data. When unfolding a ^{252}Cf spectrum using MAXIET to calculate the initial guess spectra and SPUNIT as the unfolding algorithm, Johnson found the SAN4 and UTA4 response matrices provided

results with the least error and best agreement [23:8]. For reasons presented later, a response matrix collapsed from the same origins as UTA4 is the only one used in AFITBUNKI.

SPUNIT Unfolding Algorithm

The response of a set of Bonner spheres may be written

$$C_j = \int_{E_{min}}^{E_{max}} R_j(E) \Phi(E) dE \quad j = 1, 2, \dots N \quad (1)$$

where C_j is the count rate in the j^{th} detector, $R_j(E)$ is the response of the j^{th} detector as a function of neutron energy, $\Phi(E)$ is the neutron fluence rate as a function of energy, and N is the total number of detectors. Equation (1), formally known as a Fredholm integral equation of the first kind, can not be solved in closed form in this application because $R_j(E)$ is not a continuous analytical function. It can, however, be approximated by a system of linear equations as

$$C_j = \sum_{k=1}^M R_{jk} \Phi_k \quad j = 1, 2, \dots N \quad (2)$$

where R_{jk} is the response of the j^{th} detector to neutrons in the k^{th} energy interval multiplied by the width of the k^{th} energy interval, and M is the number of energy intervals. R_{jk} is called the response matrix. If $M < N$, the system is over-determined but a non-unique solution can be found. If $M \geq N$ and both are relatively small (~ 10), a unique though not very resolved solution exists. If, as most often is the case, $M \gg N$, the system

is under-determined; error always exists and there is no unique solution. However, *a priori* knowledge can be applied if the shape of a neutron fluence spectrum is anticipated and a solution can be found which fits the estimation with minimal error.

The SPUNIT algorithm was selected as the unfolding routine for use in AFITBUNKI. Equation 3 is the basis of SPUNIT; it is a very efficient recursion formula devised by Doroshenko *et al.* [4:299]:

$$\Phi_k^{i+1} = \frac{\Phi_k^i}{\sum_j \frac{R_{jk}}{C_j^0}} \sum_j \frac{R_{jk}}{C_j^i} \quad (3)$$

C_j^0 is the measured counts in the j^{th} detector, C_j^i is calculated counts in the j^{th} detector during the i^{th} iteration, and Φ_k^i is the fluence of the k^{th} energy interval during the i^{th} iteration.

The unfolding process starts by assuming a solution for Φ_k based on *a priori* knowledge. The detector counts for the assumed solution are calculated using Equation (2). Equation (3) is then solved for each energy interval, k , by using the assumed solution (Φ_k^i), the measured detector counts (C_j^0), and the calculated detector counts (C_j^i) so that a new solution, Φ_k^{i+1} , is found. The process is iterated until a convergence criteria is met [20:3]. In BUNKI, the criteria is one of three conditions. Either the calculated average error of the fit successfully reaches what user specified, or it converges rapidly to some value other than that specified, or the specified maximum number of iterations is reached.

BUNKIUT

BUNKIUT was developed in 1986 at The University of Texas at Austin. At that time there were no major changes to the code except to convert it from FORTRAN IV to FORTRAN 77 so it could be compiled and executed on a personal computer.

Additionally, a FORTRAN routine was added by S. Peterson to allow the user to plot the unfolded spectra [10:--].

III. Development of AFITBUNKI

AFITBUNKI was developed with a 54-group energy binning structure which is closely aligned to that identified in ISO Standard 8529. In order to implement the new binning structure, Hertel and Davidson's 171 energy group response matrix, selected for its accuracy, had to be collapsed appropriately. Additionally, bin-averaged values of both the fluence for the seven initial-guess spectra and the fluence to dose conversion factors had to be calculated.

Description of AFITBUNKI

AFITBUNKI unfolds neutron fluence spectra using the measurements of up to 13 ^6LiI Bonner sphere detectors. The measured fluence spectrum is binned into 54 energy groups between 10^{-11} and 14.92 MeV. The energy end-points of the 54 groups correspond to 55 of the energy end-points of the 171-energy-group response matrix, but were chosen so they also correspond roughly to the 52 energy groups between 4.14×10^{-7} and 15 MeV ISO Standard 8529 [19:--]. In addition to those 52 energy groups, AFITBUNKI has a thermal group which extends down to 10^{-11} MeV from 4.14×10^{-7} MeV and one of the wider groups between 20 eV and 40 eV was divided into two groups. Figure 1 compares the ISO, AFITBUNKI, and BUNKI/BUNKIUT binning structures.

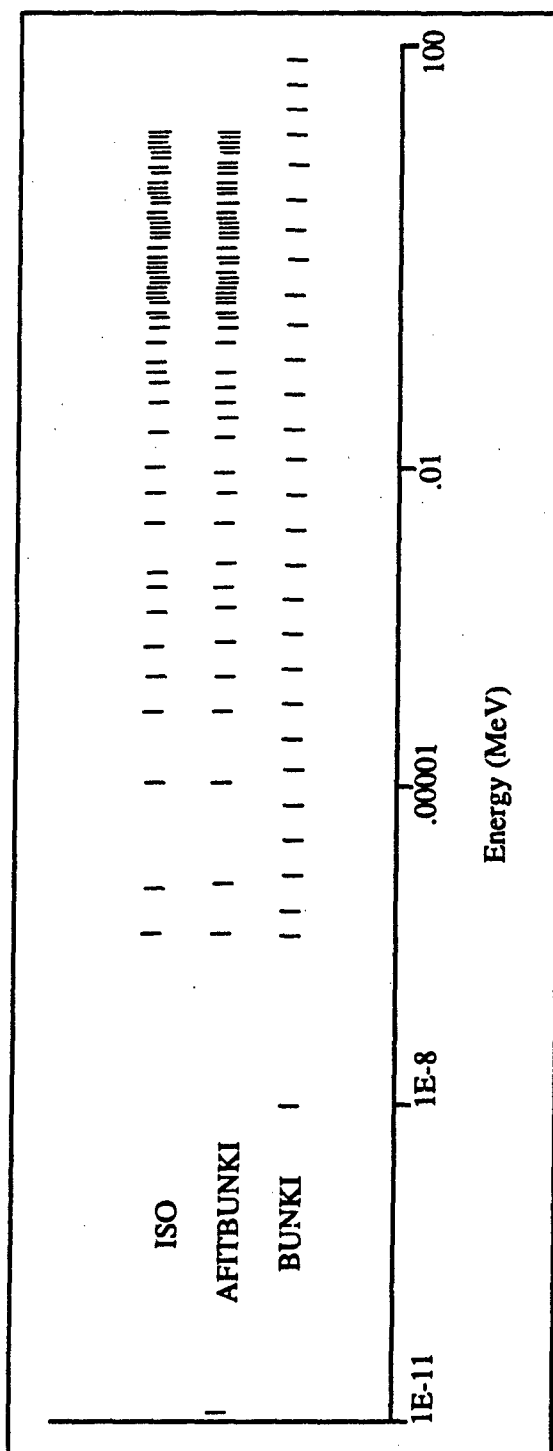


Figure 1. A number line of the energy-group endpoints comparing the binning structure used by BUNKI/BUNKIUT and AFTBUNKI and the binning structure recommended by ISO Standard 8529.

As pointed out in Johnson's study, the SAN4 and UTA4 response matrices produced the "best" results [22:543]. For this reason, Hertel and Davidson's 171 energy group response matrix which was the basis of UTA4 over the energy range of interest, was selected for use in AFITBUNKI. It was calculated using a more geometrically correct description of the ^6LiI detector, a more current cross section set, and it includes responses to neutrons with energies down to 10^{-11} MeV. The SAN4 response matrix was not selected for use in AFITBUNKI because of the concern that error would be introduced if the 25 energy groups between 4.14×10^{-7} and 15.0 MeV of SAN4 were mathematically expanded to the 54 energy groups of interest and because the SAN4 response matrix did not extend as far into the thermal region as does Hertel and Davidson's. None of the other response matrices assessed in Johnson's study were considered because of their inaccuracies [22:543].

Collapsing the 171 Energy-Group Response Matrix

Hertel and Davidson's response matrix is a calculated set of 171 energy-group neutron responses between 10^{-11} and 17.3 MeV. The responses are for a bare 4 mm x 4 mm right cylinder ^6LiI detector and for the detector enclosed in polyethylene moderators with diameters of 5.08, 7.62, 12.7, 20.32, 25.4, 30.48, 38.1, and 45.72 cm (2, 3, 5, 8, 10, 12, 15, and 18 in). The matrix includes responses to a cadmium-covered detector and cadmium-covered polyethylene moderators with diameters of 5.08, 7.62, and 12.7 cm. The response matrix was calculated with the ENDF/B-IV neutron cross-section library of DLC-41/VITAMIN-C using an adjoint transport technique [9:509; 5:315]. The energy groups of AFITBUNKI were selected such that the end-point of each group had a

corresponding end-point in the 171-group response matrix. This prevents having to interpolate a split response and introduce unnecessary error. The new 54-group response matrix collapsed from Hertel and Davidson's 171-group response matrix and used in AFITBUNKI is called UTA54.

To develop the method of collapsing the response matrix, the definition of a response matrix must be understood. A response matrix is a probability distribution function of detector counts about an incident neutron energy. A response matrix contains a table of values, each of which correspond to the average detector response per neutron entering the detector for a specified detector and specified energy interval. For any one detector, the average response, \bar{R} , in an energy interval is given by

$$\bar{R}_{\Delta E} = \frac{\int_{E_{min}}^{E_{max}} R(E) \Phi(E) dE}{\int_{E_{min}}^{E_{max}} \Phi(E) dE} \quad (4)$$

where $R(E)$ is the continuous response function and $\Phi(E)$ is the neutron fluence in the energy interval from E_{min} to E_{max} (ΔE). Since a continuous expression for R cannot be found and in order to collapse a response matrix, Equation (4) can be approximated by

$$\bar{R}_j = \frac{\sum_i^{n_i} \left[\bar{R}_i \int_{E_i}^{E_{i+1}} \Phi(E) dE \right]}{\int_{E_j}^{E_{j+1}} \Phi(E) dE} \quad (5)$$

where \bar{R}_i is the average response in each of the subintervals to be collapsed, \bar{R}_j is the average response in the collapsed (therefore larger) energy interval, and n_i is the number

of energy intervals in the uncollapsed matrix which make up the new energy interval extending from E_j to E_{j+1} . If it is assumed that the neutron fluence, $\Phi(E)$, is proportional to $1/E$, then the Equation (5) may be restated as follows:

$$\bar{R}_j = \frac{\sum_{i=1}^{n_i} \left[\bar{R}_i \int_{E_i}^{E_{i+1}} \frac{1}{E} dE \right]}{\int_{E_j}^{E_{j+1}} \frac{1}{E} dE} \quad (6)$$

The relationship used to collapse Hertel and Davidson's 171-energy group response matrix into the AFITBUNKI 54-group response matrix, UTA54, is arrived at by completing the integrals in Equation (6):

$$\bar{R}_j = \frac{\sum_{i=1}^{n_i} \bar{R}_i \log \frac{E_{i+1}}{E_i}}{\log \frac{E_{j+1}}{E_j}} \quad (7)$$

Figure 2 graphically illustrates the collapsing of several responses into a single response as given by Equation (7). Figure 3 is a plot of Hertel and Davidson's response matrix (171 energy groups) and Figure 4 is a plot of the collapsed response matrix, UTA54 (54 energy groups). A listing of UTA54 can be found in Appendix B.

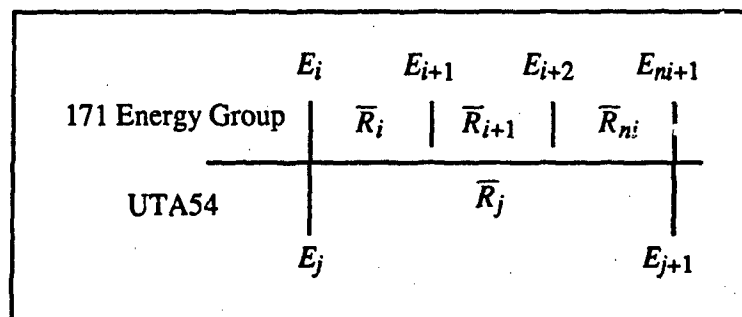


Figure 2. An illustration of the collapsing of several responses into a single response as given by Equation (7). The i subscripts refer to the energy-group endpoints and the responses of the subintervals to be collapsed and the j subscripts refer to those of the collapsed.

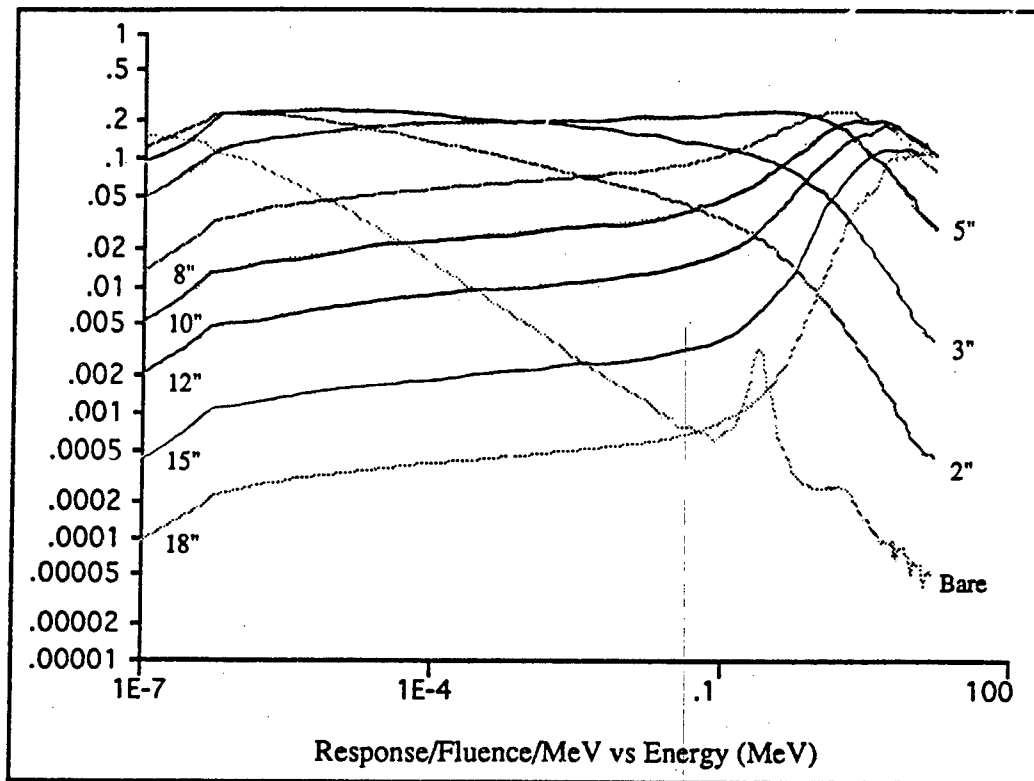


Figure 3. Hertel and Davidson's Response Matrix with 171 energy-group responses for each detector. Responses are plotted against the maximum energy of each bin. Responses to the four cadmium covered detectors not shown [9:509]

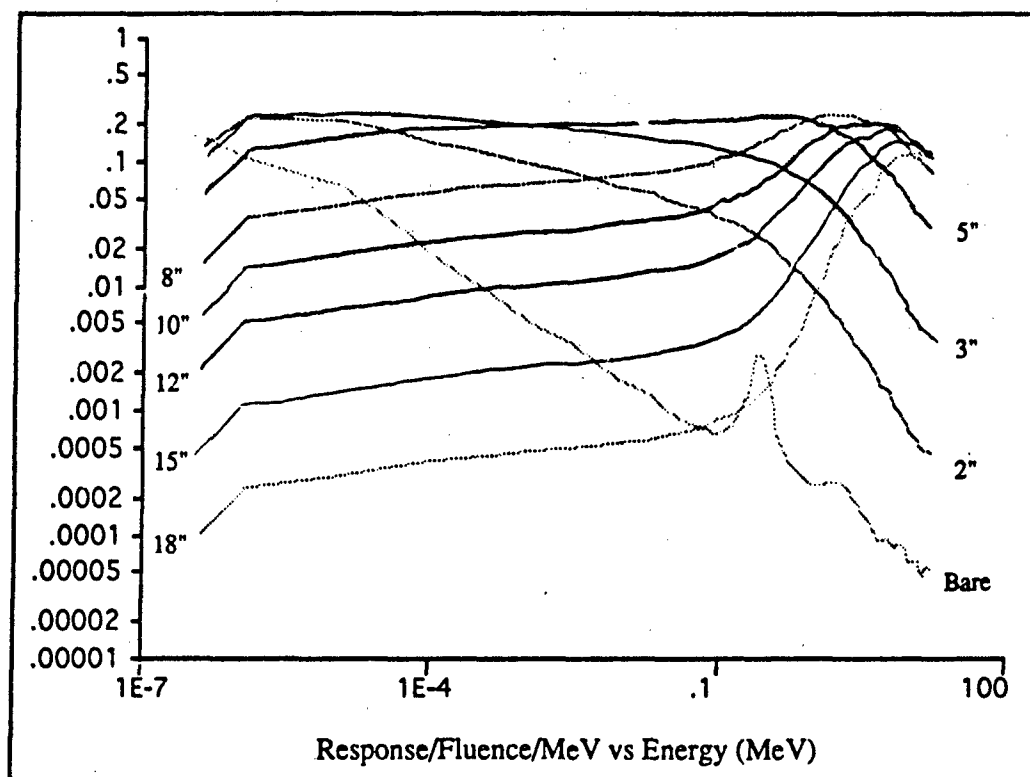


Figure 4. The UTA54 Response Matrix (54 energy-group responses) collapsed from the 171 energy-group responses for each detector for use in AFITBUNKI. Responses are plotted against the maximum energy of each bin. Responses to the four cadmium covered detectors not shown.

Initial Guess Spectra Added

The speed and precision with which AFITBUNKI converges to an acceptable solution relies on the user's *a priori* knowledge of the observed neutron spectrum. AFITBUNKI works most effectively if the user supplies an initial-guess spectrum (or starting spectrum) which has characteristics of the spectra to be unfolded. To this end, seven user-selectable initial-guess spectra have been incorporated in AFITBUNKI; alternatively, the user can calculate a combination Maxwellian plus a 1/E spectrum using the MAXIET algorithm, or can manually enter an initial-guess spectrum, or can retrieve a previously stored unfolded spectrum for use as a starting spectrum.

The seven user-selectable initial-guess neutron spectra are: flat, bare ^{252}Cf , D_2O -moderated ^{252}Cf , ^{252}Cf room return, deuterium-tritium (D-T), americium-beryllium (Am-Be), and deuterium-beryllium (D-Be). AFITBUNKI requires initial-guess spectra values to be specified as bin-averages in each of the 54 bins and the unit of the specified initial spectrum is fluence per unit lethargy (neutron/cm²/u). Output that would be meaningfully plotted from AFITBUNKI is a fluence per unit lethargy spectrum so using such a spectrum as an initial guess provides a convenient way of turning the output around for use as a starting spectrum.

The flat initial-guess spectrum assumes an equal fluence per unit lethargy in each bin. It is included in AFITBUNKI because it is the simplest and most generic starting spectrum that can be used. In AFITBUNKI, the flat spectrum has values in each bin set arbitrarily at 1 neutron/cm²/u.

The bare ^{252}Cf initial-guess spectrum was calculated from a National Institute of Standards and Technology (NIST) analytical fit which describes a ^{252}Cf neutron fluence probability distribution (neutron/cm²/MeV) up to 20 MeV. A starting spectrum is calculated by integrating the probability distribution function over all energies in each bin and dividing by the lethargy width of the bin as shown in Equation (8):

$$\Phi_j = \frac{\int_{E_j}^{E_{j+1}} \Phi(E) dE}{\log \frac{E_{j+1}}{E_j}} \quad (8)$$

The D₂O-moderated ^{252}Cf spectrum was taken from MCNP calculations provided by N.E. Hertel of the University of Texas, Austin [10:--]. He provided bin fluence values for the binning structure identified in ISO Standard 8529. Each bin was divided by its lethargy providing a fluence/lethargy spectrum. A cubic interpolation of Log(Energy)-(Φ/u) was performed to find a fluence/lethargy value at each bin's endpoint; the initial spectrum's bin-average values were found by calculating the logarithmic average of the fluence evaluated at the bin endpoints.

Initial-guess spectra (fluence/lethargy) for ^{252}Cf Room Return, D-T Fusion, Am-Be, and D-Be were also provided by N.E. Hertel. These values corresponded to the binning structure used in BUNKI and so again, a cubic interpolation of Log(Energy)-(Φ/u) was performed to find a fluence/lethargy value at each bin's endpoint; the initial spectrum's bin-average values were found by calculating the logarithmic average of the fluence evaluated at the bin endpoints.

The MAXIET algorithm was developed by T.L. Johnson *et al.* at the Naval Research Laboratory for the following reason:

“Most neutron spectra encountered in radiation protection work are produced by the scattering, moderation, and absorption of neutrons originally produced by nuclear fission, particle accelerators, or by the α, n reaction from radioactive sources. These processes tend to produce spectra that can be characterized as having a high energy peak corresponding, or reduced somewhat by moderation, to the original neutron source energy, a $(1/E)^X$ intermediate energy component produced by elastic scattering, and a thermal peak whose magnitude is determined by the atomic number of the shielding and scattering material, and by the thermal neutron absorption cross-section of these materials.” [21:4]

The MAXIET algorithm prompts the user for a temperature to calculate a “MAXIET” spectrum which has a Maxwellian high energy peak, a $1/E$ intermediate component, and a lower peak in the most thermal bin. In AFITBUNKI the MAXIET spectrum is calculated at each of the 55 AFITBUNKI energy bin endpoints. After the endpoint values are determined, bin-average fluence/lethargy values are found by calculating the logarithmic average of the bin endpoint values.

Dose and Dose Equivalent Conversion Factors

AFITBUNKI calculates the neutron fluence for each energy interval and sums the fluences over all intervals to determine the total energy-integrated fluence. By applying appropriate conversion factors, dose and dose equivalent quantities are calculated by AFITBUNKI. The dosimetric quantities and conversion factors incorporated in AFITBUNKI comply with ICRU and ICRP recommendations. Some of the conversion factors are specified directly in ICRU and ICRP publications; others have been extracted from published sources. AFITBUNKI calculates dosimetric quantities by summing the product of the conversion factor and the bin fluence in each of the 54 energy intervals.

Absorbed dose, the most fundamental dosimetric quantity, is defined by the ICRP as the energy absorbed per unit mass. It is recognized that "equal absorbed doses of radiations of different qualities may produce effects which differ." [15:2] The quantity dose equivalent accounts for this inequality and is given by $H=DQN$ where N is a modifying factor currently assigned the value unity. The quality factor, Q , varies with the effectiveness of different types of ionizing radiation and is a function of the collision stopping power in water at the point of interest. Appendix 6, Table 4 of *ICRP Publication 21* tabulates values of fluence per dose equivalent and the effective quality factor, \overline{Q} , as a function of the initial neutron energy [13:52]. Since the table in the report provides conversion factors only down to energies of 2.5×10^{-8} MeV, the conversion factors at that energy were used at 10^{-11} MeV. AFITBUNKI energy bin end-point values for H/Φ were determined from a cubic interpolation of a table of $\text{Log}(\text{Energy})$ - $\text{Log}(\Phi/H)$ values. Similarly, values for \overline{Q} for each energy bin end-point were used with these values of Φ/H to calculate end-point values of D/Φ . The logarithmic average value of

each bin's end-points for both D/Φ and H/Φ were then calculated for use as conversion factors in AFITBUNKI.

The effective dose equivalent, H_E , was first defined in *ICRP Publication 26* (1977) and is the sum of the weighted mean organ dose equivalents [14:--]. *ICRP Publication 51* (1987) tabulates values of effective dose equivalent versus energy for various irradiation geometries such as Anterior-Posterior (AP) and Rotational (Rot); however, the tabulated data are based on quality factors before they were changed [16:32]. Quality factors were tentatively redefined in 1985 as twice their previous value. In 1990, *ICRP Publication 60* (1990) definitively redefined quality factors as a function of unrestricted LET in water [IP60:81]; however, the conversion factors for H_E incorporated in AFITBUNKI are based on quality factors as defined prior to 1985. AFITBUNKI energy bin end-point values for H_E/Φ (AP) and H_E/Φ (Rot) were determined from a cubic interpolation of the tabulated (pre-1985) $\text{Log}(E)$ - $\text{Log}(H_E/\Phi)$ values. Again, since the table in the report only provides conversion factors down to energies of 2.5×10^{-8} MeV, the conversion factors at that energy were used again at 10^{-11} MeV. The logarithmic average value of each bin's end-points for both H_E/Φ (AP) and H_E/Φ (Rot) was then calculated for use as conversion factors in AFITBUNKI.

Effective dose equivalent, H_E is the current standard for measuring the dose equivalent resulting from wholebody exposure. However, coincident with the 1990 redefining of the quality factor was the introduction of a new dosimetric quantity, effective dose, E . The ICRP intends for effective dose to replace effective dose equivalent but this is not yet the case. H_E (AP) and H_E (Rot) are presented in AFITBUNKI because they are the current standard and contrast well with the newly

defined values of effective dose, E .

When the ICRP redefined quality factors in *ICRP Publication 60*, it was felt that the absorbed dose modified by the new quality factor relationship did not accurately reflect the probability of detriment resulting from exposure because of uncertainties in the radiobiological information [IP60:81]. Therefore, a radiation weighting factor, based on the type and energy of radiation, was introduced. Additionally, a new relationship for equivalent dose to tissue, H_T , was introduced. When this quantity is modified by a tissue-importance weighting factor and summed up over all tissues/organs, the effective dose, E , is found.

ICRP Publication 60 tabulates radiation and tissue weighting factors [IP60:82-86] from which J. Tanner at Pacific Northwest Laboratory calculated fluence-to-effective dose conversion factors (E/Φ) for neutrons with energies between 0.001 and 10 MeV. incident in various geometries on an anthropomorphic phantom; see Table 1. Iso refers to isotropic exposure. AFITBUNKI energy bin end-point values for E/Φ (AP) and E/Φ (Iso) were determined from a cubic interpolation of the tabulated $\text{Log}(\text{Energy})$ - $\text{Log}(E/\Phi)$ values shown in Table 1. Since the data only extends down to 10^{-3} MeV, the conversion factors at that energy were used again at 10^{-11} MeV. Similarly, the values reported at 10 MeV were used at 15 MeV. The logarithmic average value of each bin's end-points for both E/Φ (AP) and E/Φ (Rot) was then calculated for use as conversion factors in AFITBUNKI.

The calculation of effective dose equivalents and the determination of particular organ doses is impractical in the case of radiation workers exposed to external radiation sources. However, when individuals are exposed to ionizing radiation, it is useful to

Table 1
FLUENCE-TO-EFFECTIVE DOSE CONVERSION FACTORS

Neutron Energy, MeV	$E/\Phi(\text{AP})$ pSv-cm ²	$E/\Phi(\text{Iso})$
.001	14.52	6.75
.010	18.42	8.27
.020	24.02	11.25
.050	40.60	18.49
.100	62.26	25.93
.200	99.90	41.25
.500	188.42	77.42
1.000	254.74	106.99
2.000	389.93	189.33
3.000	435.71	230.28
4.000	475.50	253.28
5.000	472.28	278.66
6.000	477.23	286.62
8.000	481.15	303.99
10.000	481.24	309.05

Fluence-to-effective dose conversion factors for neutrons incident on an Anthropomorphic Phantom in the Anterior-Posterior and Isotropic geometries [10:--].

specify the dose numerically. The ICRU has defined the operational radiation protection quantity ambient dose equivalent, H^* , for environmental monitoring. This quantity gives an adequate approximation to the effective dose equivalent from strongly penetrating external sources and is defined in ICRU Reports 39 (1985) and 43 (1988) [17:3; 18:4]. ICRU Report 43 defines it as follows:

"The ambient dose equivalent, $H^*(d)$, at a point in a radiation field, is the dose equivalent that would be produced by the corresponding aligned and expanded field, in the ICRU sphere at depth, d , on the radius opposing the direction of the aligned field." [18:4]

The ICRU sphere is a 30 cm tissue equivalent sphere, and the recommended depth, d , is 10 mm.

Since ambient dose equivalent is meant as an approximation of H_E from survey meter results, it is based on quality factors as opposed to radiation and tissue weighting factors. AFITBUNKI calculates ambient dose equivalent based on both pre-1985 (*ICRP Publication 26*-based) and post-1985 (*ICRP Publication 60*-based) quality factors. *ICRP Publication 60* states that the ICRU will be examining ambient dosimetric quantities in detail as part of a general revision of *ICRP Publication 51* (1987) which will incorporate the new radiation weighting factors, and therefore a new definition is imminent [IP60:88].

The fluence-to-ambient dose, $D^*(10)/\Phi$, and pre-1985, fluence-to-ambient dose equivalent, $H^*(10)/\Phi$, conversion factors implemented in AFITBUNKI are calculated using the analytical fit of S.R. Wagner *et al.* [28:232]. The post-1985 fluence-to-ambient dose equivalent, $H^*(10)/\Phi$, conversion factors were calculated using the analytical fit of H. Schuhmacher *et al.* [27:86]. These fits are specified for neutrons of energies between 2.5×10^{-8} and 20 MeV so the values at 2.5×10^{-8} again were used at 10^{-11} MeV. AFITBUNKI calculates the ambient quality factor, Q^* , as the ratio of $H^*(10)/D^*(10)$ for both *ICRP Publication 26*-based and *ICRP Publication 60*-based values. A sample listing of AFITBUNKI output is listed in Appendix C.

IV. Experimental Data

Well characterized neutron sources were used to verify and validate AFITBUNKI. Those chosen include bare and D₂O-moderated ²⁵²Cf neutron sources, recommended by the ISO as reference radiation fields for calibrating neutron measuring devices [19:--], and a polyethylene-moderated ²⁵²Cf neutron source which has a significant thermal neutron component. The highly thermalized spectrum allows testing of the effect of the UTA54 binning and the accuracy and validity of unfolding spectra at such low energies. The detector counts, corrected for scattered neutrons, may also be modified by detector calibration factors. Two data sets for each source, one modified by calibration factors and the other not, were unfolded by both AFITBUNKI and BUNKIUT using various parameters and the results compared.

Room Return Correction

Air- and room-scattered neutrons have a lower average energy than a source spectrum and need to be considered in calibration and measurement situations; hence in experiments where the direct neutron source strength is required, the Bonner sphere detector data must be corrected for neutrons scattered by the air and the surfaces in the room in which the measurement is taken. Scatter corrections can be made computationally by modeling the source, room, and detector using Monte Carlo methods, or experimentally by using shadow shields to directly measure the inscatter

contributor , or simplified analytical models developed to fit the experimental data.

Monte Carlo methods are straightforward but can require a detailed model and a significant amount of CPU time. The shadow-shield method involves blocking the line-of-sight path between the neutron source and the detector with a strong neutron absorber. The difference in detector measurements made with and without the shield in place is the correction for neutrons in-scattered by air and the room surfaces. This method can be difficult since shadow shields tend to be rather large and heavy and the detector, shield, and source are usually several meters above the floor to minimize scattered neutron effects.

The technique attributed to Eisenhower, Schwartz, and Johnson [3:43] and described by J.B. Hunt relies on observing the perturbation from the spherical divergence of the neutron flux. As a detector is moved away from the source, the count-rate, C , should drop off as the inverse square of the distance, R , between the source and detector. A plot of $(C \times R^2)$ versus R^2 should yield a straight line with no slope if there are no scattered neutrons influencing the count. If the scattered neutron contribution is significant, the slope of the line, $(C \times R^2)$ versus R^2 , will have a positive slope. If a number of measurements made at various values of R are plotted, a least-squares fit of the data can be made and extrapolated down to an R^2 point where the scattered neutron contribution is insignificant. This intercept provides the direct neutron source strength at one meter [11:239].

Calibration Factors for UTA4

Bonner sphere calibration factors are applied because response matrices are calculated assuming a detector efficiency of 1.0. Since detectors are not 100% efficient, each detector should be normalized experimentally and a calibration factor defined. Calibration factors are determined by measuring the flux from a well-characterized source, using the response matrix to unfold the spectrum, and then comparing the measured flux to the calculated flux of the source. In theory, once a detector system is calibrated to a response matrix, the calibration factors should hold for any source. In practice this is not always the case and it might be speculated that calibration factors may vary with average neutron energy. In any case, calibration factors are generally close to unity and may be applied to each Bonner sphere detector, or a single value may be applied to the system as a whole. Some researchers ignore them altogether; however, from the results presented here, calibration factors defined for use with BUNKIUT's UTA4 response matrix have been found to be useful information when unfolding spectra and they have been applied to the data unfolded by AFITBUNKI using the UTA4 response matrix.

Bare ^{252}Cf Source Spectra

Bare ^{252}Cf , a well-characterized neutron source recommended by the ISO as a reference radiation field, was used to verify and validate AFITBUNKI. The bare ^{252}Cf Bonner sphere detector response data (ball data) used in this work was collected by R.S. Hartley in 1987 [7:--]. The data has been corrected for room return and air-scattered neutrons. The geometry of the ^{252}Cf source (capsule type SR-CF-100) and detector system is described in detail in Reference 6. Table 2 contains a summary of the bare ^{252}Cf detector responses used in this work.

Table 2

BONNER SPHERE DETECTOR DATA FROM A BARE ^{252}Cf SOURCE

Bonner Sphere Detector, Diameter <u>[in. (cm)]</u>		Unmodified Data <u>(counts)</u>	Calibration Factor <u>(no units)</u>	Modified Data <u>(counts)</u>
2	(5.08)	3.15	1.18	2.788
3	(7.62)	17.397	1.09	15.961
5	(12.7)	60.021	1.05	57.16
8	(20.32)	72.956	1.05	69.48
10	(25.4)	56.45	1.02	55.34
12	(30.48)	38.18	1.01	37.80
18	(45.72)	11.448	1.09	10.503

All data has been corrected for room return and air-scattered neutrons. The detector "name" corresponds to its size in inches. The modified data has been corrected using the calibration factor presented in the table. The standard error in the 2 inch data is 4.7%; all others are less than 2%.

D₂O-Moderated ²⁵²Cf Source Spectra

D₂O-moderated ²⁵²Cf, a well-characterized neutron source recommended by the ISO as reference radiation field, was used to verify and validate AFITBUNKI. The D₂O-moderated ²⁵²Cf Bonner sphere detector response data used in this work was collected by N.E. Hertel in 1991 [10:--]. The ²⁵²Cf source was centered in a 15 cm radius sphere wrapped with a 0.5 mm thick layer of cadmium and filled with heavy water as recommended by the NIST [8:23]. Table 3 summarizes the D₂O-moderated ²⁵²Cf detector response unfolded in this work. The data has been corrected for room return

Table 3

BONNER SPHERE DETECTOR DATA FROM A D₂O-MODERATED ²⁵²Cf SOURCE

Bonner Sphere Detector, Diameter <u>[in. (cm)]</u>	Unmodified Data <u>(counts)</u>	Calibration Factor <u>(no units)</u>	Modified Data <u>(counts)</u>
bare	4.48	1.11	4.03
2 (5.08)	32.17	1.18	27.19
3 (7.62)	54.85	1.02	53.66
5 (12.7)	66.74	1.02	65.69
8 (20.32)	36.48	1.005	36.30
10 (25.4)	21.62	1.01	21.39
12 (30.48)	12.31	0.973	12.65
18 (45.72)	3.39	1.00	3.39

All data has been corrected for room return and air-scattered neutrons. The detector "name" corresponds to its size in inches. Modified data has been corrected using the calibration factors obtained by Hertel for D₂O-moderated ²⁵²Cf. The standard errors in the data are estimated to be less than 2%.

and air-scattered neutrons. It should be noted that the calibration factors provided by Hertel for his measurements differ from those provided by Hartley. Since the D_2O -moderated ^{252}Cf source produces a softer neutron spectrum, a response from the bare 6LiI detector appears significant.

Polyethylene-Moderated ^{252}Cf Source Spectra

Polyethylene-moderated ^{252}Cf neutron sources produce a significant thermal neutron component which allows testing of the effect of re-binning to increase thermal sensitivity. The polyethylene-moderated ^{252}Cf Bonner sphere detector response data used in this work was collected by R.S. Hartley in 1987 [7:--]. The ^{252}Cf source was centered in a 22.86 cm radius polyethylene sphere with a cavity that was 0.5 cm in diameter. Table 4 summarizes the polyethylene-moderated ^{252}Cf detector responses unfolded in this work. Since the polyethylene-moderated ^{252}Cf source produces more thermalized neutrons, a response from the bare ^6LiI detector would have been desirable; however, this data was not collected and is not used in this case.

Table 4

BONNER SPHERE DETECTOR DATA FROM A POLYETHYLENE-MODERATED ^{252}Cf SOURCE

Bonner Sphere Detector Size <u>[in. (cm)]</u>	Unmodified Data <u>(counts)</u>	Calibration Factor <u>(no units)</u>	Modified Data <u>(counts)</u>
2 (5.08)	1.540	1.18	1.305
3 (7.62)	1.906	1.09	1.749
5 (12.7)	2.629	1.05	2.504
8 (20.32)	2.337	1.05	2.226
10 (25.4)	1.803	1.02	1.768
12 (30.48)	1.353	1.09	1.241

All data has been corrected for room return and air-scattered neutrons. The detector "name" corresponds to its size in inches. Modified data has been corrected using the calibration factor presented in the table. The standard errors in the data are calculated to be less than 5.5%.

Analyses of Bonner Sphere Data with AFITBUNKI and BUNKIUT

A bare ^{252}Cf spectrum, a D_2O -moderated ^{252}Cf spectrum, and a polyethylene-moderated ^{252}Cf spectrum have been unfolded using ball data with and without calibration factors applied. In each case, two different spectra were used as initial guesses. Additionally, there are three other parameters which must be specified when using AFITBUNKI or BUNKI: a) the maximum root-mean-square error in the final spectrum fit (the end-test specified in percent), b) the number of iterations between testing for the fit error, and c) the maximum number of iterations before terminating the run if no tolerable fit is calculated. The most sensitive of these is the number of iterations between end-tests; it is selected subjectively based only on experience acquired by unfolding numerous spectra. It is the most sensitive because a solution may converge to the specified maximum root-mean-square error in half or two-thirds of the specified number of iterations between end-tests, but allowing the iterations to continue helps to sufficiently smooth the final solution.

All of the spectra unfolded by BUNKI used the UTA4 response matrix and the SPUNIT unfolding algorithm.

In unfolding all three spectra, regardless of the combination of the starting parameters, it was found that if calibration factors were not applied, it may require anywhere from two to twenty times as many iterations to converge to a solution with the same end-test as obtained by unfolding data which had been modified by calibration factors. Even then, total fluence and average energy values varied unpredictably. For this reason, all data used in this research was modified by calibration factors.

The bare ^{252}Cf ball data were unfolded using a flat and a bare ^{252}Cf starting spectrum. In each case the specified end-test was 1%, the number of iterations between end-tests was 100, and the maximum number of iterations allowed was 5000.

Both the D_2O -moderated ^{252}Cf ball data and the polyethylene-moderated ^{252}Cf ball data were unfolded using a flat and a MAXIET spectrum as the starting spectra. The specified end-test was 1%, the number of iterations between end-tests was 500, and the maximum number of iterations allowed was 5000. In the case of the MAXIET starting spectra, a temperature of 1.1 MeV was specified.

Comparison of AFITBUNKI and BUNKIUT Results

Bonner sphere detector data were unfolded by both AFITBUNKI and BUNKI and the results compared to verify that AFITBUNKI was developed properly. There are two ways to compare the AFITBUNKI results with the BUNKIUT results. The first is a comparison of energy-integrated quantities. The most important of these is the total energy-integrated fluence since the primary objective of the code is to iteratively unfold fluence spectra. By summing the unfolded spectrum convoluted with dosimetric conversion factors over all energies, energy-integrated dosimetric quantities are calculated and can be compared as well. AFITBUNKI and BUNKIUT are similar enough that they should produce results of at least the same order of magnitude and more likely only differing by a few percent. The second method is to compare the energy-dependent fluence spectra unfolded by each code. Since the primary difference between calculating fluence spectra with AFITBUNKI and BUNKIUT is the energy binning

structure, it is expected that the spectra should have generally the same shape but the AFITBUNKI-calculated spectra would show the finer structuring which occurs at the higher energies.

As stated earlier, bare ^{252}Cf and D_2O -moderated ^{252}Cf spectra are well characterized and recommended for verification and validation purposes. The polyethylene-moderated ^{252}Cf responses were unfolded primarily to demonstrate the effect of extending the range of the most thermal bin down to 10^{-11} MeV from 10^{-8} MeV and to show the difference in spectra as unfolded by AFITBUNKI and BUNKIUT in that range.

Table 5 is a comparison of the AFITBUNKI and BUNKIUT results of the unfolded bare ^{252}Cf detector responses. It shows that regardless of the starting spectrum, AFITBUNKI and BUNKIUT produce results that differ by less than a few percent. The results in the table also verify that the conversion factors for dose and dose equivalent implemented in AFITBUNKI are correct inasmuch as they agree with the BUNKIUT values.

Table 5

**AFITBUNKI AND BUNKIUT RESULTS OF UNFOLDED
DETECTOR RESPONSES FROM A BARE ^{252}Cf SOURCE**

<u>Code</u>	<u>Start Spectrum</u>	<u>Error (%)</u>	<u>Number of Iters</u>	<u>Total Fluence (n/cm²)</u>	<u>Average Energy (MeV)</u>	<u>Dose (pGy)</u>	<u>Dose Equiv (pSv)</u>	<u>Effective Quality Factor</u>
AFITB	Flat	0.73	100	337.2	2.21	1.23+4	1.13+5	9.21
BUNKI	Flat	0.65	100	337.3	2.17	1.21+4	1.12+5	9.24
AFITB	Bare Cf	0.72	100	336.8	2.07	1.21+4	1.11+5	9.19
BUNKI	Bare Cf	0.69	100	337.2	2.06	1.19+4	1.10+5	9.22

All values are for 2, 3, 5, 8, 10, 12, and 18 inch ball data corrected for room return and calibration factors before unfolding. The number of iterations between end-tests was 100 and the end-test was 1%. Average energy does not include thermal groups below 0.414 eV. Dose equivalent values are based on quality factors as defined by *ICRP Publication 21* for comparison purposes.

Figures 5 and 6 are plots of the unfolded bare ^{252}Cf spectra using the two different starting spectra. The spectra have been normalized to 1 source neutron/cm².

Comparisons of the plotted spectra are subjective but show that they generally agree well; the finer binning structure of AFITBUNKI is apparent over the range shown. The unfolded spectra appear to agree regardless of starting spectra used.

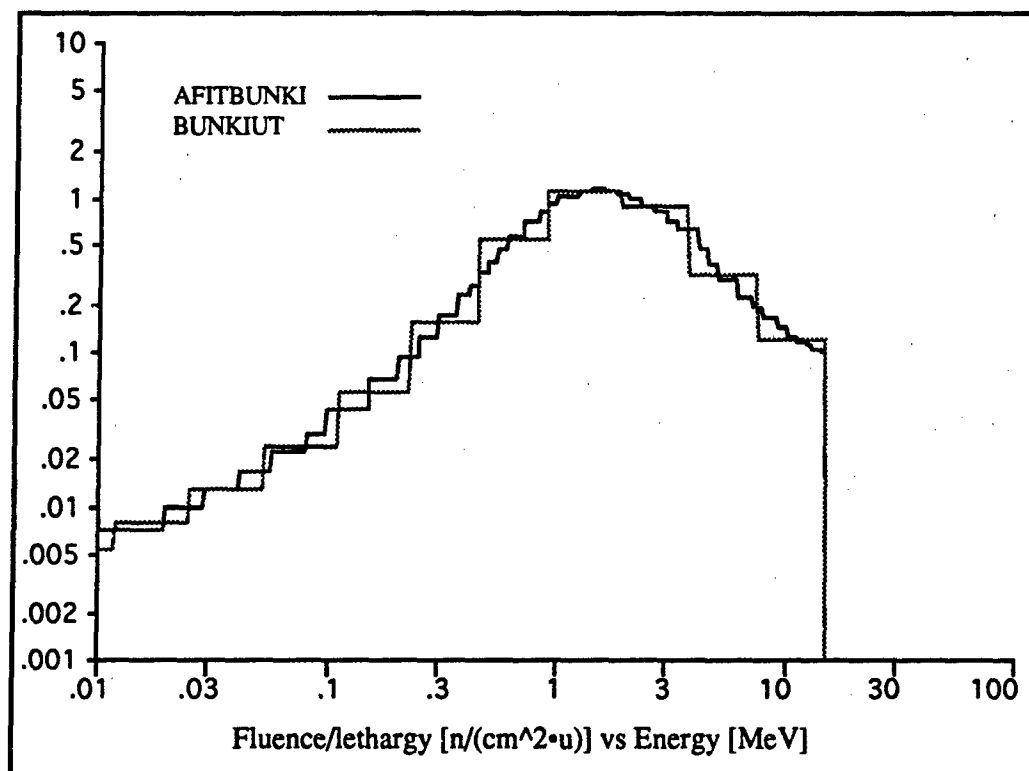


Figure 5. Bare ²⁵²Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a flat starting spectrum. The spectra are normalized to 1 source neutron/cm².

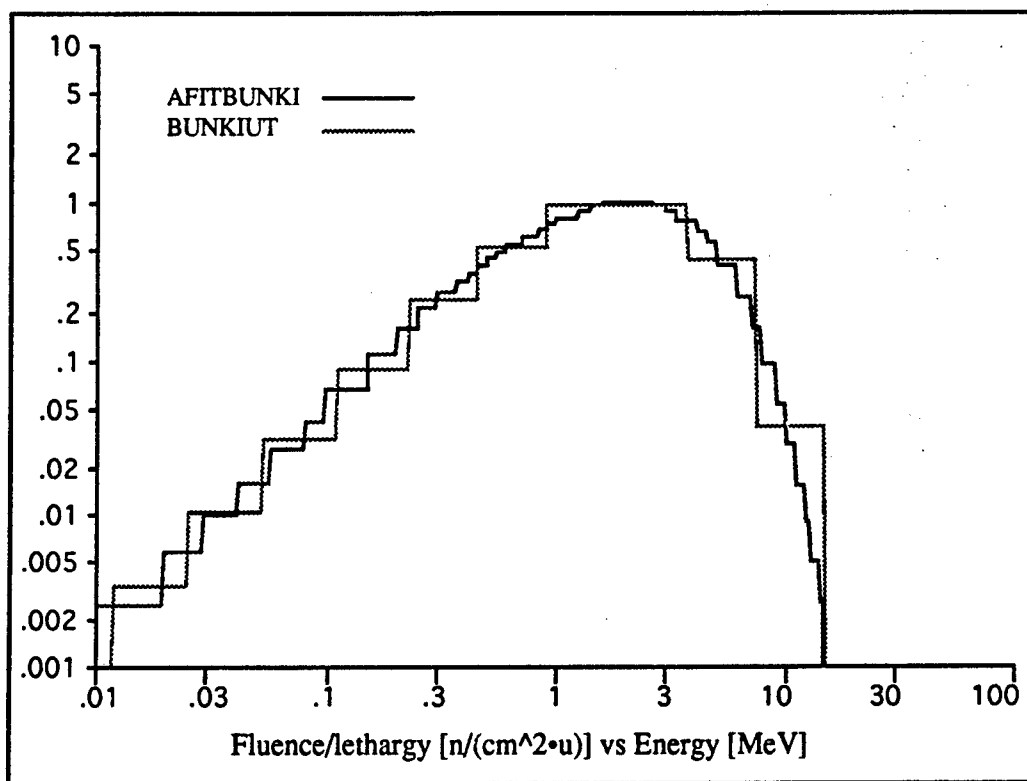


Figure 6. Bare ^{252}Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a bare ^{252}Cf starting spectrum. The spectra are normalized to 1 source neutron/cm².

Table 6 is a comparison of the AFITBUNKI and BUNKIUT results of the unfolded D₂O-moderated ²⁵²Cf ball data. In this case, the table shows that for a given starting spectrum, AFITBUNKI and BUNKIUT again produce total fluence, average energy, dose, and dose equivalent values that agree within several percent.

Table 6

AFITBUNKI AND BUNKIUT RESULTS OF UNFOLDED
DETECTOR RESPONSES FROM A D₂O-MODERATED ²⁵²Cf SOURCE

<u>Code</u>	<u>Start Spectrum</u>	<u>Error (%)</u>	<u>Number of Iters</u>	<u>Total Fluence (n/cm²)</u>	<u>Average Energy (MeV)</u>	<u>Dose (pGy)</u>	<u>Dose Equiv (pSv)</u>	<u>Effective Quality Factor</u>
AFITB	Flat	0.30	500	357.2	0.755	4.84+3	3.40+4	7.02
BUNKI	Flat	0.24	500	356.8	0.752	4.97+3	3.39+4	6.82
AFITB	MAXIET	0.48	500	350.2	0.505	4.60+3	3.31+4	7.19
BUNKI	MAXIET	0.42	500	350.0	0.513	4.61+3	3.28+4	7.11

All values are for bare, 2, 3, 5, 8, 10, and 12 inch ball data corrected for room return and calibration factors before unfolding. The number of iterations between end-tests was 500 and the end-test was 1%. Average energy does not include thermal groups below 0.414 eV. Specified temperature in calculating MAXIET spectrum was 1.1 MeV. Dose equivalent values are based on quality factors as defined by ICRP Publication 21 for comparison purposes.

Figures 7 and 8 are plots of the unfolded D_2O -moderated ^{252}Cf spectra using the two different starting spectra. The spectra have been normalized to 1 source neutron/cm². Here too, comparisons of the plotted spectra are subjective but show that they generally agree. The unfolded D_2O -moderated ^{252}Cf spectra are more complex than the bare ^{252}Cf spectra and the agreement is not quite so good in these cases. Notable in the plots is AFITBUNKI's larger binning structure at the lower energies.

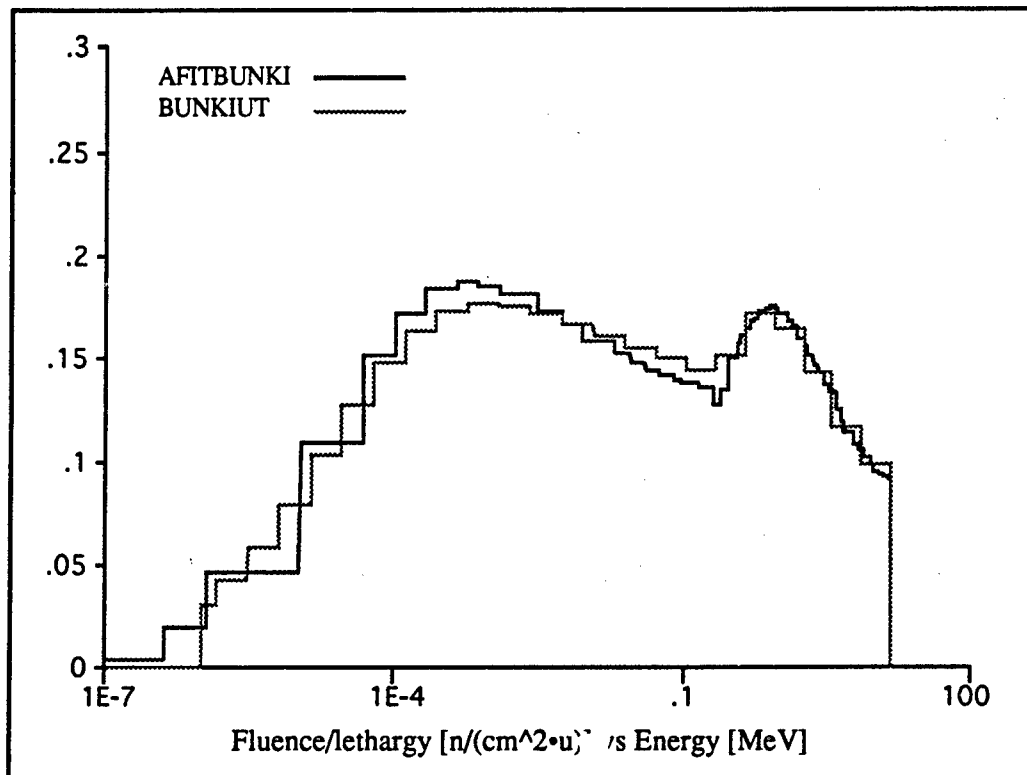


Figure 7: D_2O -moderated ^{252}Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a flat starting spectrum. The spectra are normalized to 1 source neutron/cm².

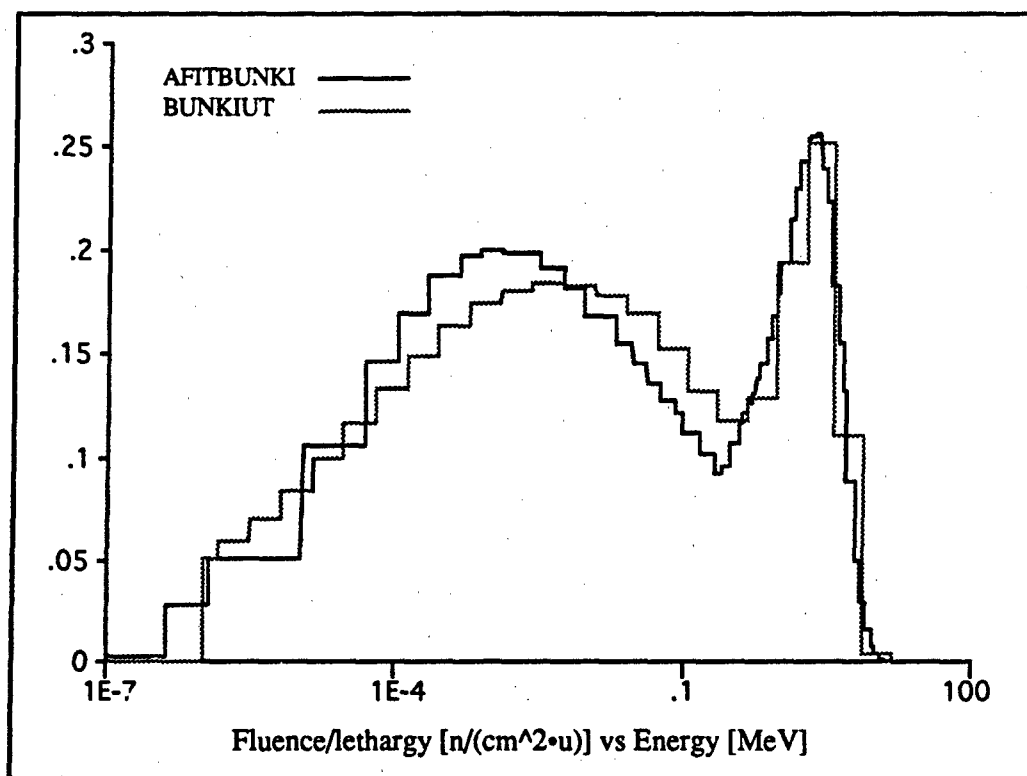


Figure 8. D_2O -moderated ^{252}Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a MAXIET starting spectrum. The specified temperature in calculating the MAXIET spectrum was 1.1 MeV. The spectra are normalized to 1 source neutron/cm².

Table 7 is a comparison of the AFITBUNKI and BUNKIUT results of the unfolded polyethylene-moderated ^{252}Cf ball data. It is obvious from the table that when unfolding the spectra, both AFITBUNKI and BUNKIUT reached the specified number of maximum iterations (5000) before converging to a solution which met the end-test value of 1%. Nonetheless, for a given starting spectrum, AFITBUNKI and BUNKIUT produce total fluence, dose and dose equivalent values which agree within 6%. Again, this verifies that AFITBUNKI produces reasonable results.

Table 7

AFITBUNKI AND BUNKIUT RESULTS OF UNFOLDED DETECTOR
RESPONSES FROM A POLYETHYLENE-MODERATED ^{252}Cf SOURCE

<u>Code</u>	<u>Start Spectrum</u>	<u>Error (%)</u>	<u>Number of Iters</u>	<u>Total Fluence (n/cm²)</u>	<u>Average Energy (MeV)</u>	<u>Dose (pGy)</u>	<u>Dose Equiv (pSv)</u>	<u>Effective Quality Factor</u>
AFITB	Flat	1.05	5000	19.66	2.23 (1.53)	4.43+3	3.51+4	7.92
BUNKI	Flat	1.10	5000	18.79	2.30	4.47+3	3.50+4	7.83
AFITB	MAXIET	1.01	5000	19.65	2.00 (1.35)	4.32+3	3.43+4	7.95
BUNKI	MAXIET	1.04	5000	19.16	2.21	4.38+3	3.44+4	7.85

All values are for 2, 3, 5, 8, 10, and 12 inch ball data corrected for room return and calibration factors before unfolding. The number of iterations between end-tests was 500 and the end-test was 1%. Average energy does not include thermal groups below 0.414 eV. Average energy values in parentheses include all energy groups. Specified temperature in calculating MAXIET spectrum was 1.1 MeV. Dose equivalent values are based on quality factors as defined by *ICRP Publication 21* for comparison purposes.

Figures 9 and 10 are plots of the unfolded spectra. The spectra are normalized to 1 source neutron/cm². Again, comparisons of the plotted spectra are subjective but show

that they generally agree everywhere except in the most thermal bin. In that bin, the lethargy fluence measured by BUNKIUT is approximately two times greater than the lethargy fluence measured by AFITBUNKI. However, the fluence in the most thermal bin as measured by BUNKIUT is approximately 10% less than the fluence as measured by AFITBUNKI. This is likely an artifact of the technique used by Johnson to collapse Hertel and Davidson's 171 energy-group response matrix. It is probable that the responses for energies between 10^{-11} MeV and 10^{-8} MeV were incorporated into the UTA4 response matrix's lowest bin so as not to underestimate the total fluence of a highly thermalized neutron spectrum. Since dosimetric conversion factors appear to be fairly constant at low energies, Johnson's method would provide a reasonable estimate of the quantities when integrated over all energies.

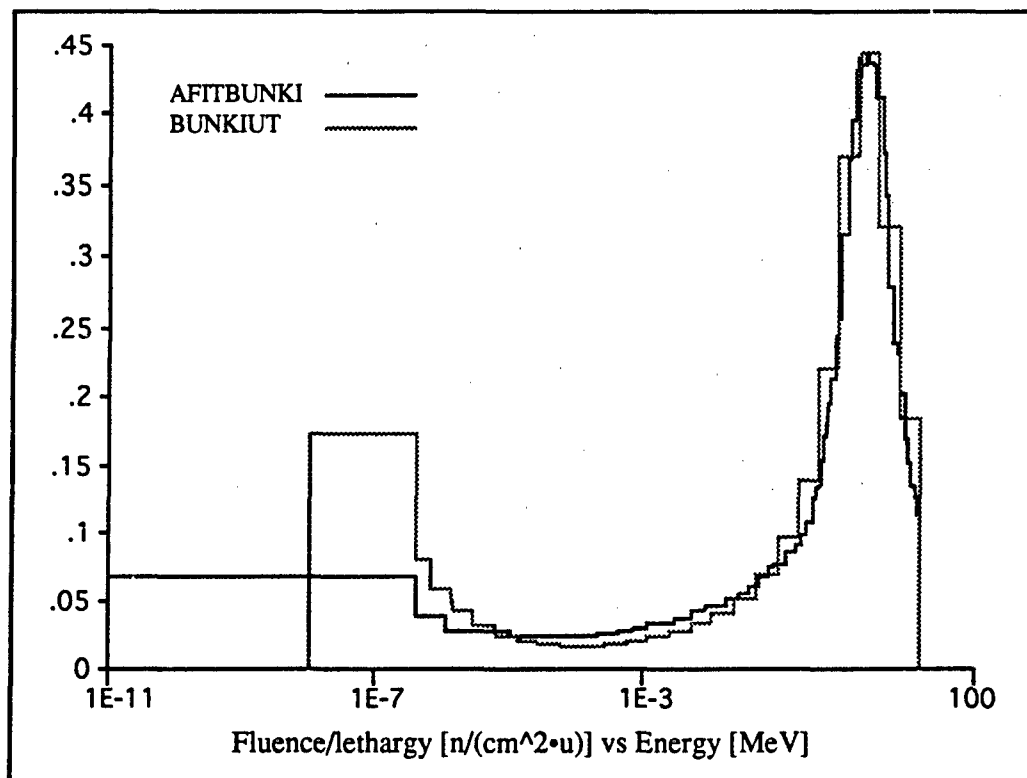


Figure 9. Polyethylene-moderated ^{252}Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a flat starting spectrum. The spectra are normalized to 1 source neutron/cm².

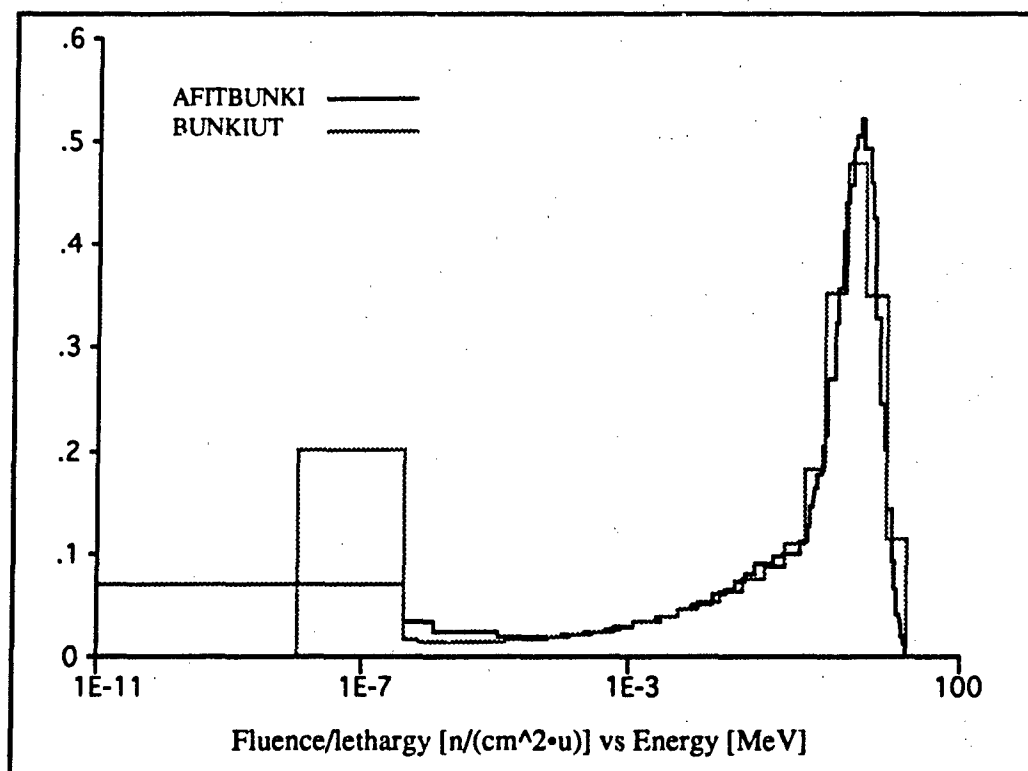


Figure 10. Polyethylene-moderated ^{252}Cf Spectrum unfolded by AFITBUNKI and BUNKIUT using a MAXIET starting spectrum. The specified temperature in calculating the MAXIET spectrum was 1.1 MeV. The spectra are normalized to 1 source neutron/cm².

V. Calculated Responses

In order to validate AFITBUNKI, reference spectra need to be selected and responses calculated. Bare ^{252}Cf and D_2O -moderated ^{252}Cf fission spectra are recommended for quantitative comparisons because both are well described by the National Institute of Standards and Technology. The polyethylene-moderated ^{252}Cf spectrum was used because it has a relatively large percentage of thermal neutrons. Three quantities by which spectra may be compared independent of the total measured fluence are the average neutron energy, \bar{E} , absorbed dose-to-fluence ratio, H/Φ , and ambient dose-to-fluence ratio, H^*/Φ .

Bare ^{252}Cf - \bar{E} , H/Φ , and H^/Φ*

The NIST bare ^{252}Cf fission spectrum is described up to 20 MeV by means of a reference Maxwellian, $M(E)$, modified by four piecewise continuous segments below 6 MeV plus one exponential above 6 MeV. The reference Maxwellian is given by

$$M(E) = 0.667 \sqrt{E} e^{-1.5E/2.13} \quad (9)$$

where E is energy in MeV, and the evaluated spectrum is

$$\chi(E) = M(E) \times \mu(E) \quad (10)$$

where $\chi(E)$ is in $(\text{cm}^2\text{-MeV})^{-1}$. The adjustment functions, $\mu(E)$, are as identified in Table 8 [12:426]. A plot of the NIST analytical fit of the bare ^{252}Cf spectrum obtained from this expression and integrated over all energies is shown in Figure 11.

Table 8
NIST ADJUSTMENT FUNCTIONS FOR BARE ^{252}Cf

Energy Interval (MeV)			$\mu(E)$			
0.0	-	0.25	1.0	+	$1.20E$	- 0.237
0.25	-	0.8	1.0	-	$0.14E$	+ 0.098
0.8	-	1.5	1.0	+	$0.024E$	- 0.0332
1.5	-	6.0	1.0	-	$0.0006E$	+ 0.0037
6.0	-	20.0	$1.0 \exp[-0.03(E - 6.0)/1]$			

[12:426]

Based on the analytical fit, the average neutron energy of the spectrum was calculated to be 2.11 MeV. By using the dose equivalent and ambient dose equivalent conversion factors described in Section III to convert the spectrum to dose equivalents, an H/Φ value of 336.9 pSv-cm² was obtained. Similarly, an $H^*(10)/\Phi$ value (ICRP Publication 26-based) of 342.5 pSv-cm² and an $H^*(10)/\Phi$ value (ICRP Publication 60-based) of 449.0 pSv-cm² was obtained.

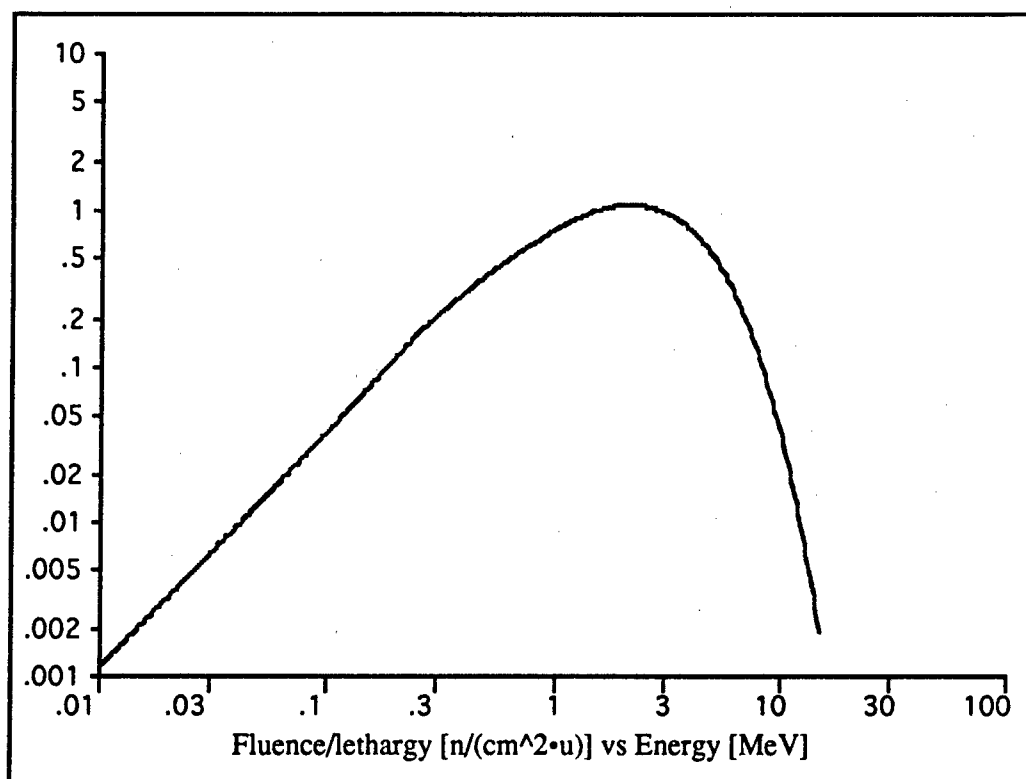


Figure 11. Bare ^{252}Cf Spectrum based on the NIST analytical fit. The spectrum is normalized to 1 source neutron / cm^2 .

D₂O-Moderated ²⁵²Cf - \bar{E} , H/Φ , and H^/Φ*

The D₂O-moderated source in use at the NIST consists of a double-walled stainless steel sphere of 15 cm radius filled with D₂O and covered with 0.5 mm thick cadmium to absorb thermal neutrons [8:23]. A source access tube with an outside diameter of 0.8 cm runs through the center of the sphere. The ²⁵²Cf source is placed inside a 0.64 cm radius iron sphere, which in turn is placed inside the D₂O-filled, 0.8 mm-thick iron access tube.

The NIST sphere has been modeled with MCNP by N.E. Hertel and J.C. McDonald [8:24] using the continuous energy ENDF/B-V neutron cross section library (RMCCS1), DLC-105. The fluences were binned into 54 energy groups similar to those described above for use in AFITBUNKI. The group structure includes the 52 ISO groups between 4.14x10⁻⁷ and 15 MeV, and two additional groups. A large thermal group extending from 10⁻¹¹ to 4.14x10⁻⁷ MeV was added and the original group between 0.02 and 0.04 MeV was split into two groups. The modeling yielded the fluence spectrum shown in Figure 12 [8:24].

Based on MCNP, the average neutron energy of the spectrum calculated by Hertel and McDonald is 0.548 MeV. Fluence to dose equivalent conversion factors interpolated from *ICRP Publication 21* were folded with the MCNP binned fluences to provide an H/Φ value of 92.9 pSv-cm². Ambient dose equivalent conversion factors suggested by Siebert and Hollnagel [26:145] were folded with the spectrum to yield an H^*/Φ value (*ICRP Publication 26-based*) of 94.9 pSv-cm² [8:24].

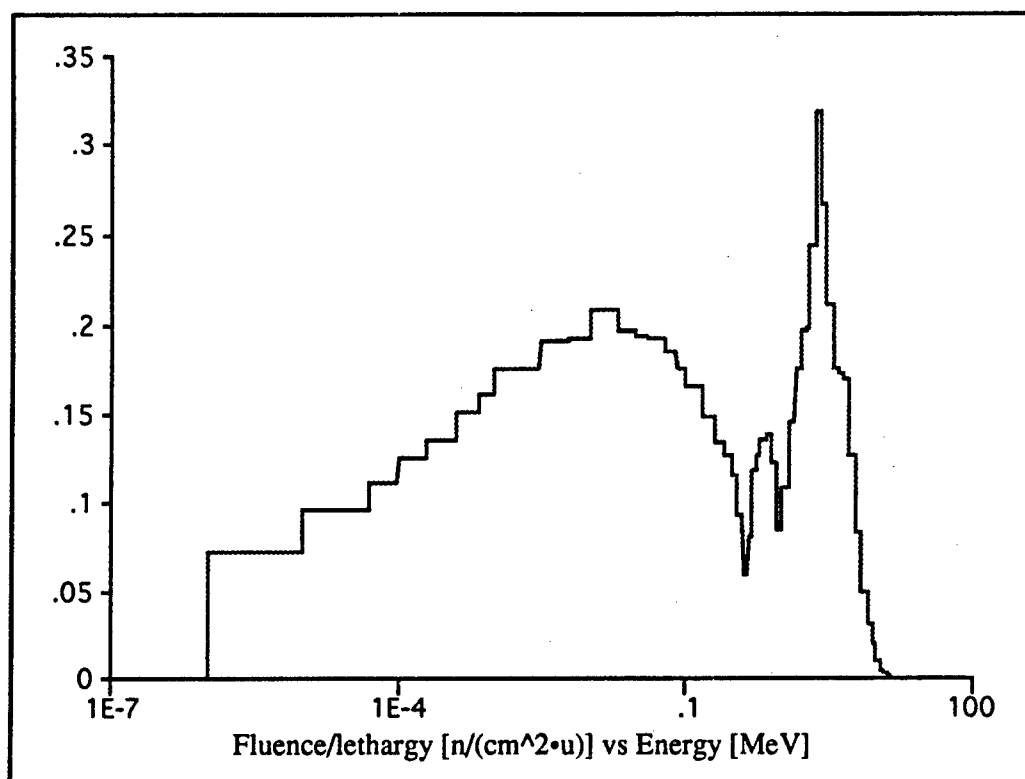


Figure 12. D₂O-moderated ²⁵²Cf Spectrum based on an MCNP model of the NIST sphere. The spectrum is normalized to 1 source neutron /cm² [8:26].

Polyethylene-Moderated ^{252}Cf - \bar{E} , H/Φ , and H^*/Φ

The polyethylene-moderated ^{252}Cf source has been modeled with MCNP at The Air Force Institute of Technology. The model simplified the geometry of the source by assuming a simple polyethylene sphere with a diameter of 45.72 cm and an inner cavity diameter of 0.5 cm. The ^{252}Cf source was assumed to be a point source located at the center of the air-filled cavity. MCNP calculates a Maxwellian fission spectrum distribution based on an input temperature of 1.466 MeV. The continuous energy ENDF/B-V neutron cross section library was used as well as thermal particle scattering $S(\alpha,\beta)$ tables which take into account the effects of chemical binding and crystal structure for neutron energies below 4 eV at room temperatures. The polyethylene (C_2H_6) sphere modeled had a density of 0.934 g/cm³; the air filled cavity was assumed to be composed of 78 atom-% nitrogen, 21 atom-% oxygen, and 1 atom-% argon with a density of 1.1833×10^{-3} g/cm³.

MCNP tallied fluences which were binned into the same 54 energy groups used in AFITBUNKI. Based on 10^5 particle histories, the total fluence calculated had a statistical error of 1.4%. The simplified geometry model yielded the fluence spectrum shown in Figure 13. From the MCNP results, the average neutron energy of the spectrum was calculated to be 1.149 MeV. By using the dose equivalent and ambient dose equivalent conversion factors described in Section III to convert the fluence spectrum to dose equivalents, an H/Φ value of 152.9 pSv-cm² was obtained. Similarly, an $H^*(10)/\Phi$ value (ICRP Publication 26-based) of 155.9 pSv-cm² and an $H^*(10)/\Phi$ value (ICRP Publication 60-based) of 201.5 pSv-cm² was obtained.

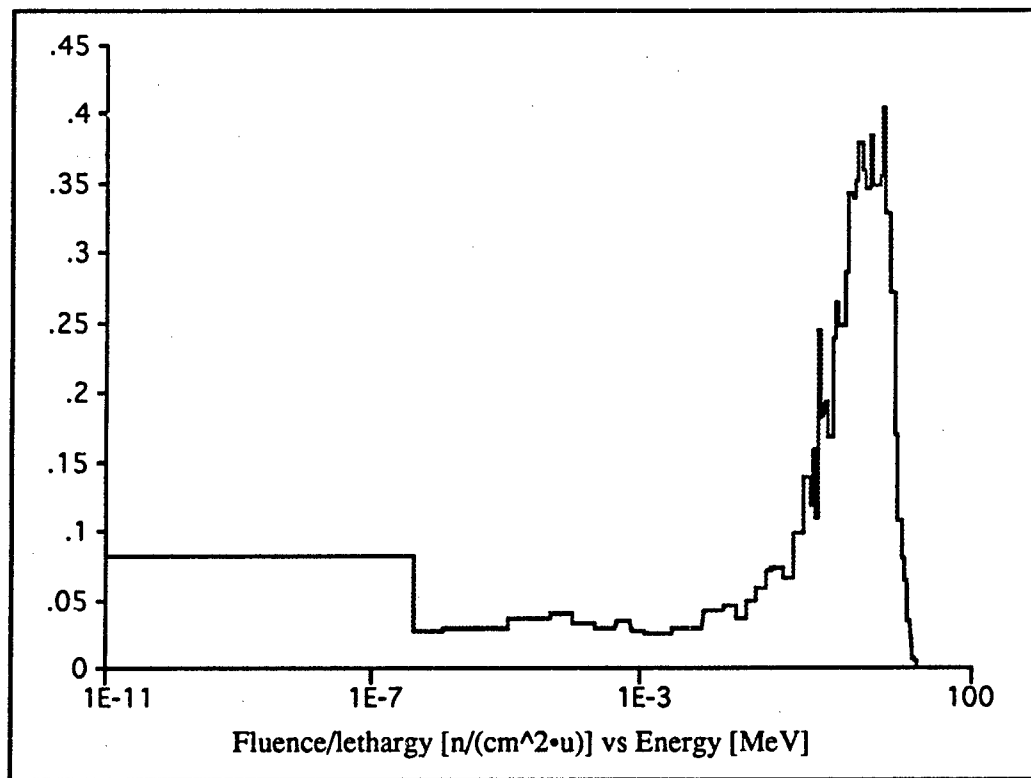


Figure 13. Polyethylene-moderated ^{252}Cf Spectrum based on an MCNP calculation of a simplified model of the source. The spectrum is normalized to 1 source neutron /cm².

VI. Validation of AFITBUNKI

Experimental vs. Calculated Results

As noted in the comparison of AFITBUNKI and BUNKIUT results, there are two ways to compare the AFITBUNKI experimental results with the calculated results. The first is to compare energy-integrated values which are independent of the total measured fluence. The values compared here are the average neutron energy, \bar{E} , absorbed dose-to-fluence ratio, H/Φ , and ambient dose-to-fluence ratio, H^*/Φ . The second method is to compare the fluence spectra as measured by AFITBUNKI with the calculated reference fluence spectra presented in Section V. Both of the above comparisons have been made for each of the reference sources.

\bar{E} , H/Φ , and H^*/Φ for Bare ^{252}Cf . Table 9 is a summary of comparison values obtained from AFITBUNKI measurements and from calculations made using the NIST analytical fit. The average energy values unfolded by AFITBUNKI differ from the NIST reference value by as much as only 5% and the three dose equivalent-to-fluence ratios all differ by less than 2%. AFITBUNKI provided the lowest error in the average energy calculation when using the bare ^{252}Cf starting spectrum but it also provided dose equivalent-to-fluence ratios with the greatest difference. In contrast, using the flat starting spectrum, AFITBUNKI unfolded dose equivalent-to-fluence ratios with the least difference but also measured the average energy with a 5% difference.

Table 9

COMPARISON OF AFITBUNKI-MEASURED AND
NIST ANALYTICAL FIT-CALCULATED VALUES
FOR BARE ^{252}Cf FISSION SPECTRUM

<u>Reference Value</u>	<u>Start Spectrum</u>	<u>Average Energy (MeV)</u>	<u>H/Φ (ICRP 21) (pSV·cm²)</u>	<u>$H^*(10)/\Phi$ (ICRP 26) (pSV·cm²)</u>	<u>$H^*(10)/\Phi$ (ICRP 60) (pSV·cm²)</u>
NIST Fit	--	2.11	336.9	342.5	449.0
<u>Present Measurements</u>					
AFITBUNKI	Flat	2.21	336.2	342.5	448.3
	Bare Cf	2.08	331.3	337.4	443.1

All AFITBUNKI values are for 2, 3, 5, 8, 10, 12, and 18 inch ball data corrected for room return and air-scattered neutrons, and modified by calibration factors. The number of iterations between end-tests was 100 and the end test was 1%. The average energy does not include thermal groups below 0.414 eV. All of the dose equivalent-to-fluence ratios were calculated based on the sources identified in Section III.

$\Phi(E)$ for Bare ^{252}Cf . Figures 14 and 15 are plots of the AFITBUNKI unfolded spectra overlayed onto the NIST fit of a bare ^{252}Cf fission spectrum. Observations of the spectra, though subjective, show all three starting spectra provide good estimates of the accepted standard. Using the flat starting spectra tended to soften the peak fluence and show greater fluence at higher energies. Subjectively, the best looking unfolded spectrum is produced by using a bare ^{252}Cf starting spectrum. The resulting spectrum (Figure 15) appears to match the reference spectrum extremely well.

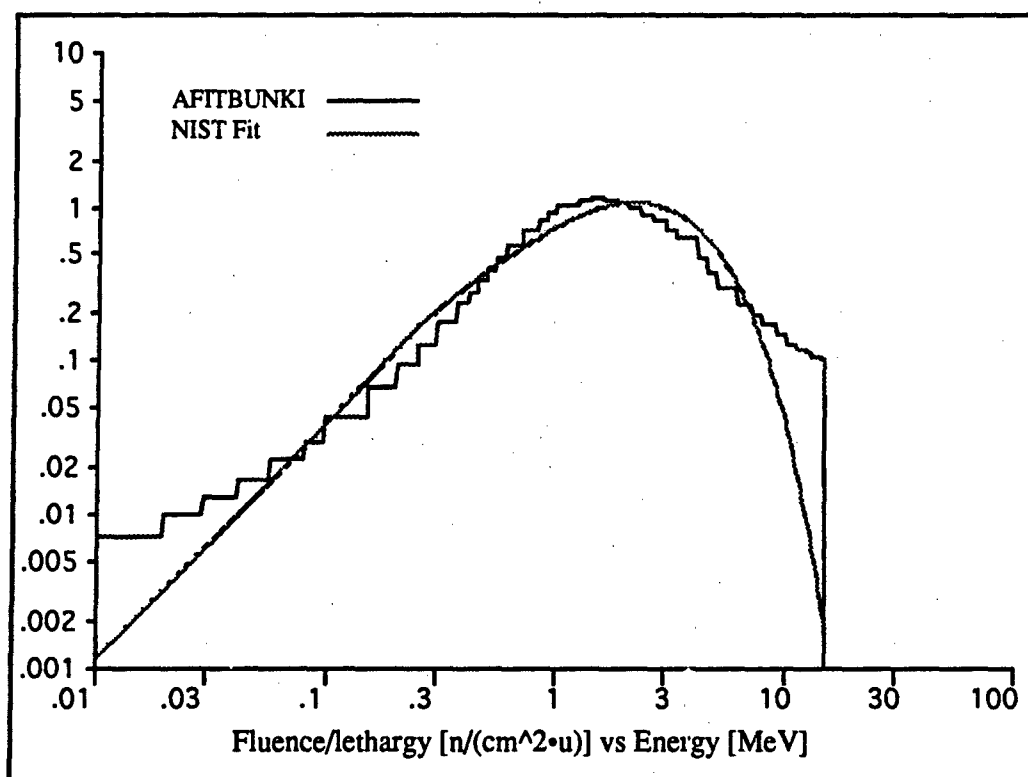


Figure 14. Bare ^{252}Cf Spectrum represented by the NIST analytical fit and unfolded by AFITBUNKI using a flat starting spectrum. The spectra are normalized to 1 source neutron $/\text{cm}^2$.

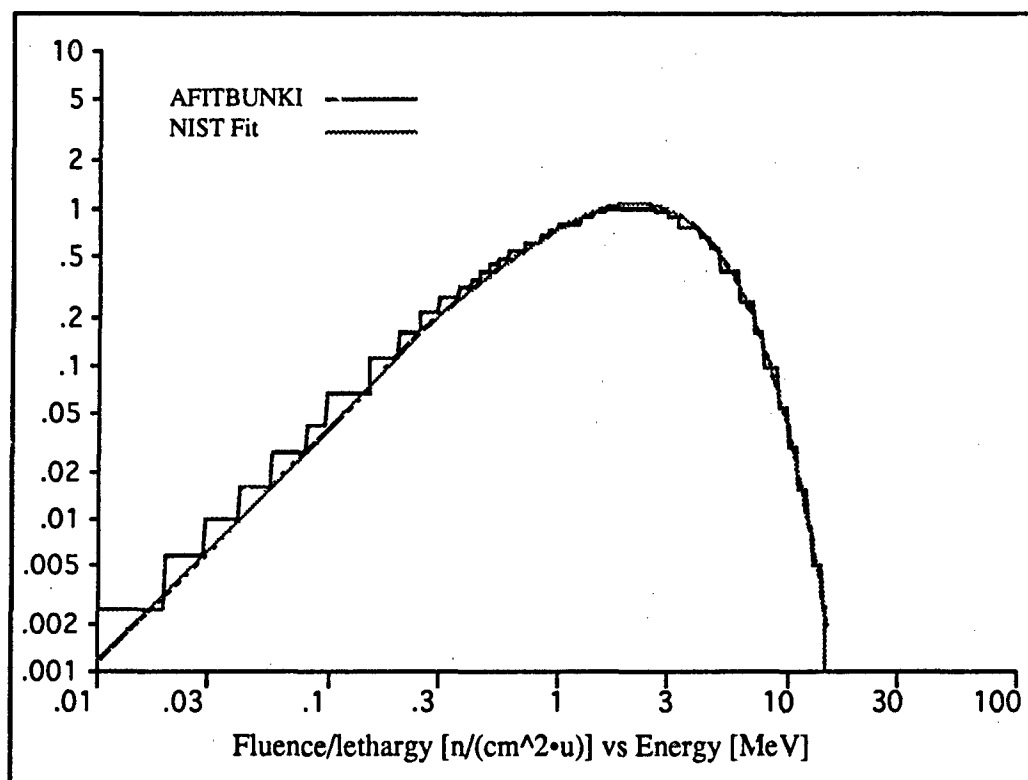


Figure 15. Bare ^{252}Cf Spectrum represented by the NIST analytical fit and unfolded by AFITBUNKI using a bare ^{252}Cf starting spectrum. The spectra are normalized to 1 source neutron / cm^2 .

\bar{E} , H/Φ , and H^*/Φ for D_2O -Moderated ^{252}Cf . Table 10 is a summary of comparison values obtained from AFITBUNKI measurements and from calculations made by N.E. Hertel and J.C. McDonald's MCNP model of the NIST D_2O -Moderated ^{252}Cf source [8:25,26]. The average energy values calculated by AFITBUNKI differ from the NIST value by as much as 39%; however, the two dose equivalent-to-fluence ratios all differ by less than 5%. Using the MAXIET starting spectrum provided the lowest difference in both the average energy (7%) and dose equivalent-to-fluence ratios (both less than 2%). This improvement over the spectrum unfolded using a flat spectrum

Table 10

COMPARISON OF AFITBUNKI-MEASURED AND
MCNP-CALCULATED VALUES FOR
 D_2O -MODERATED ^{252}Cf FISSION SPECTRUM

<u>Reference Value</u>	<u>Start Spectrum</u>	<u>Average Energy (MeV)</u>	<u>H/Φ (ICRP 21) (pSV·cm²)</u>	<u>$H^*(10)/\Phi$ (ICRP 26) (pSV·cm²)</u>
NIST Calculation of Hertel and McDonald	--	0.548	92.9	94.9
<u>Present Measurements</u>				
AFITBUNKI	Flat	0.761	95.1	100.1
	MAXIET	0.512	94.4	95.9

All AFITBUNKI values are for 2, 3, 5, 8, 10, 12, and 18 inch ball data corrected for room return and air-scattered neutrons, and modified by calibration factors. The number of iterations between end-tests was 500 and the end test was 1%. The average energy does not include thermal groups below 0.414 eV. The specified MAXIET temperature was 1.1 MeV. All of the AFITBUNKI-measured dose equivalent-to-fluence ratios were calculated based on the sources identified in Section III. All of the dose equivalent-to-fluence ratios applied to NIST MCNP model were based on the sources identified in Reference HER4.

indicates the usefulness of *a priori* information. The D_2O -moderated ^{252}Cf spectrum is fairly complex in that it has several peaks. When AFITBUNKI unfolds the spectrum using virtually informationless flat starting spectra, large discrepancies in the unfolded spectra appear. However, by using the MAXIET starting spectrum which at least contains the information of a single peak, considerably more accurate energy integrated results are achieved.

$\Phi(E)$ for D_2O -Moderated ^{252}Cf . Figures 16 and 17 are plots of the AFITBUNKI unfolded spectra overlayed onto Hertel and McDonald's NIST MCNP calculation of a D_2O -moderated ^{252}Cf fission spectrum. One of the objectives in constructing AFITBUNKI was to select an energy binning structure that was as close as possible to that identified by the ISO. It is interesting to note that the energy binning structure of the MCNP model (ISO-specified) and AFITBUNKI are close enough as to be indistinguishable on the plot.

Again, observations of the spectra are subjective but this time a little more telling. The flat starting spectrum tended to soften the peaks and group together the two higher energy ones. The MAXIET starting spectrum also provides softer peaks but because the unfolding process used more *a priori* information via the starting spectrum, the shape tends to more accurately reflect the MCNP-calculated reference.

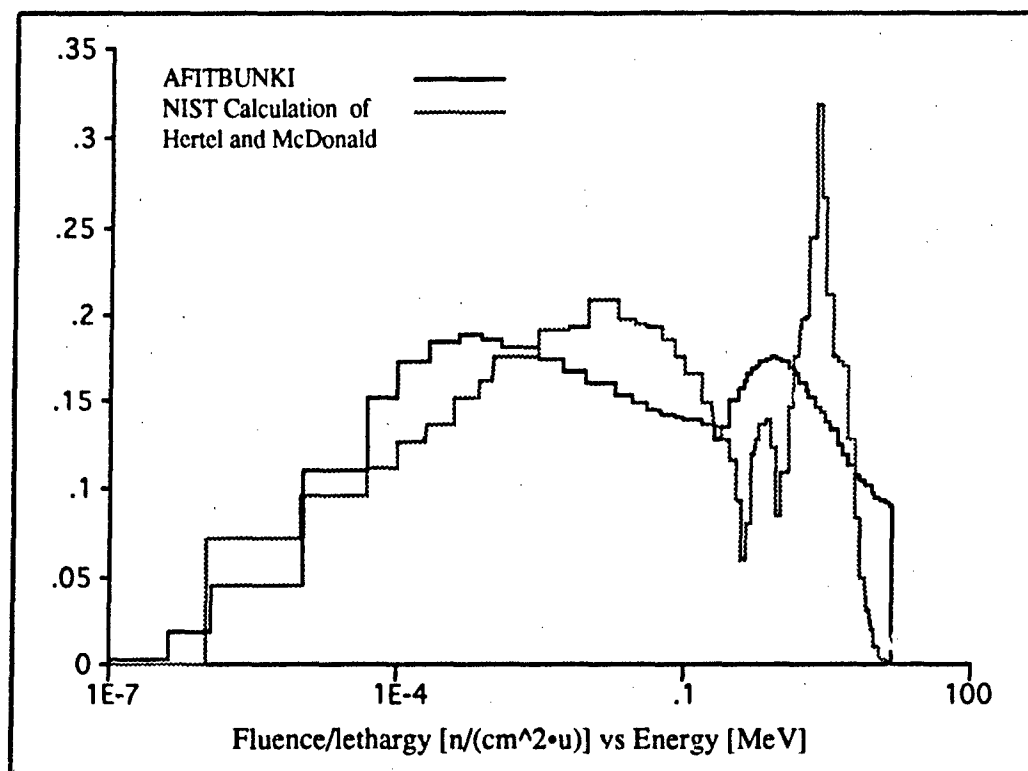


Figure 16. D₂O-moderated ²⁵²Cf Spectrum represented by Hertel and McDonald's NIST MCNP calculation and unfolded by AFITBUNKI using a flat starting spectrum. The spectra are normalized to 1 source neutron/cm².

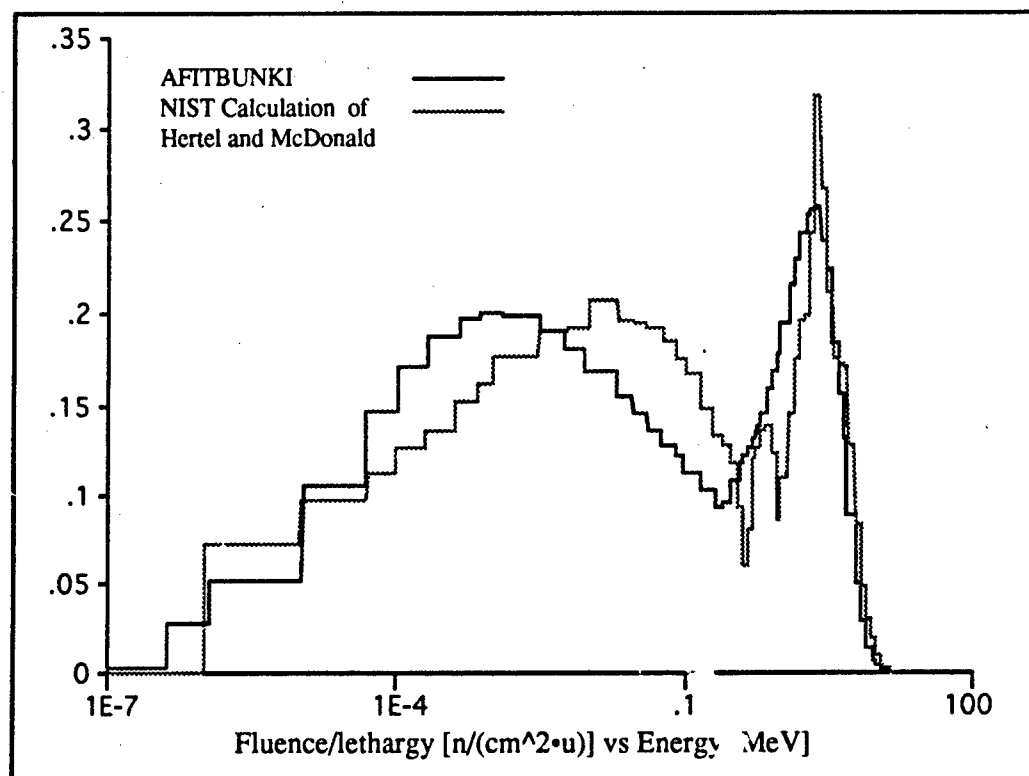


Figure 17. D₂O-moderated ²⁵²Cf Spectrum represented by Hertel and McDonald's NIST MCNP calculation and unfolded by AFITBUNKI using a MAXIET starting spectrum. The specified temperature in calculating the MAXIET spectrum was 1.1 MeV. The spectra are normalized to 1 source neutron/cm².

\bar{E} , H/Φ , and H^*/Φ for Polyethylene-Moderated ^{252}Cf . Table 11 is a summary of comparison values obtained from AFITBUNKI measurements and from conversion factors folded with MCNP fluence calculations of the polyethylene-moderated ^{252}Cf source. The average energy values calculated by AFITBUNKI differ from the NIST value by as much as 33% and as little as 17%. The three dose equivalent-to-fluence ratios all over-estimate the reference by as much as 17%. Again, using the MAXIET starting spectrum provided the lowest difference in both the average energy (17%) and dose equivalent-to-fluence ratios (all less than 14%).

Table 11

COMPARISON OF AFITBUNKI-MEASURED AND
MCNP-CALCULATED VALUES FOR POLYETHYLENE-
MODERATED ^{252}Cf FISSION SPECTRUM

<u>Reference Value</u>	<u>Start Spectrum</u>	<u>Average Energy (MeV)</u>	<u>H/Φ (ICRP 21) (pSV·cm²)</u>	<u>$H^*(10)/\Phi$ (ICRP 26) (pSV·cm²)</u>	<u>$H^*(10)/\Phi$ (ICRP 60) (pSV·cm²)</u>
MCNP Simplified Geometry Model	--	1.149	152.9	155.9	201.5
<u>Present Measurements</u>					
AFITBUNKI	Flat	1.529	178.4	182.8	232.5
	MAXIET	1.347	174.6	177.0	226.6

All AFITBUNKI values are for 2, 3, 5, 8, 10, 12, and 18 inch ball data corrected for room return and air-scattered neutrons, and modified by calibration factors. The number of iterations between end-tests was 500 and the end test was 1%. The average energy includes all energy groups. The specified MAXIET temperature was 1.1 MeV. All of the AFITBUNKI and MCNP dose equivalent-to-fluence ratios were calculated based on the sources identified in Section III.

$\Phi(E)$ for Polyethylene-Moderated ^{252}Cf . Figures 18 and 19 are plots of the AFITBUNKI unfolded spectra overlayed onto the simplified geometry MCNP calculation of a polyethylene-moderated ^{252}Cf fission spectrum.

Again, observations of the spectra are subjective. One of the reasons for comparing AFITBUNKI output with the polyethylene-moderated ^{252}Cf fission spectrum was to validate the thermal bin responses. It is interesting to note how closely the thermal bin of the AFITBUNKI spectra matches that of the MCNP spectra. This supports the speculation that when BUNKI's UTA4 response matrix was collapsed from the 171 energy-group response matrix, the response assigned to the most thermal bin, however intentional, may have been overestimated. It is also apparent from both figures why the energy-integrated values are relatively high; both the flat and MAXIET starting spectra produced unfolded spectra with peaks as much as 20% higher than the reference spectrum.

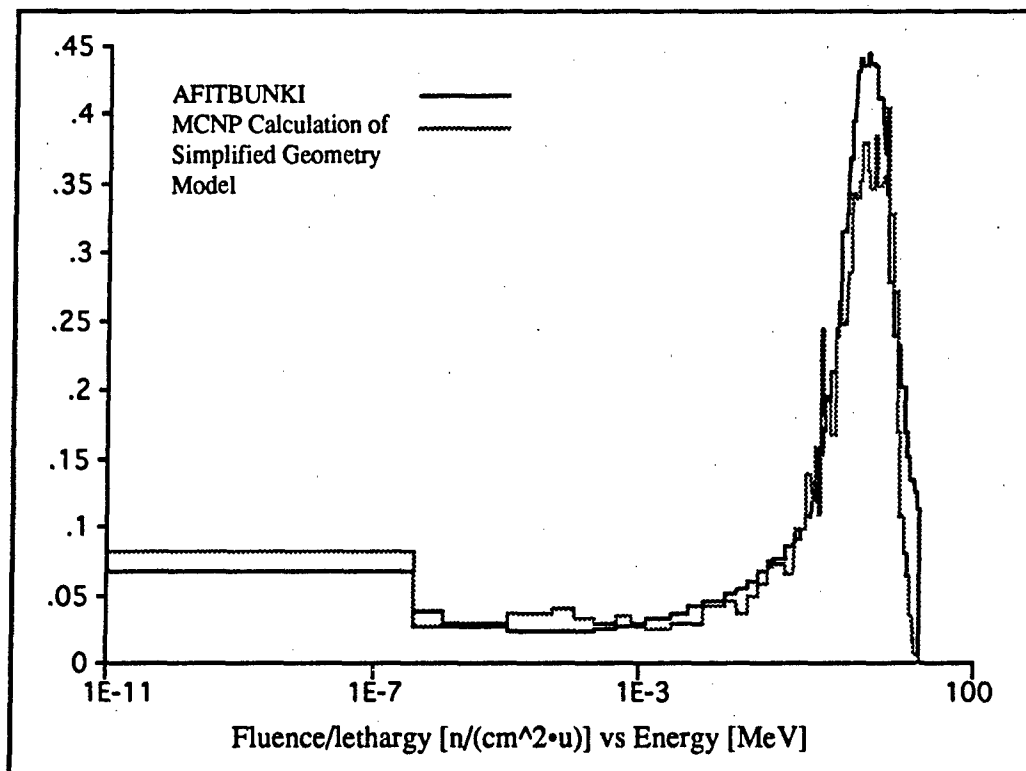


Figure 18. Polyethylene-moderated ^{252}Cf Spectrum represented by a simplified geometry MCNP calculation and unfolded by AFITBUNKI using a flat starting spectrum. The spectra are normalized to 1 source neutron/cm².

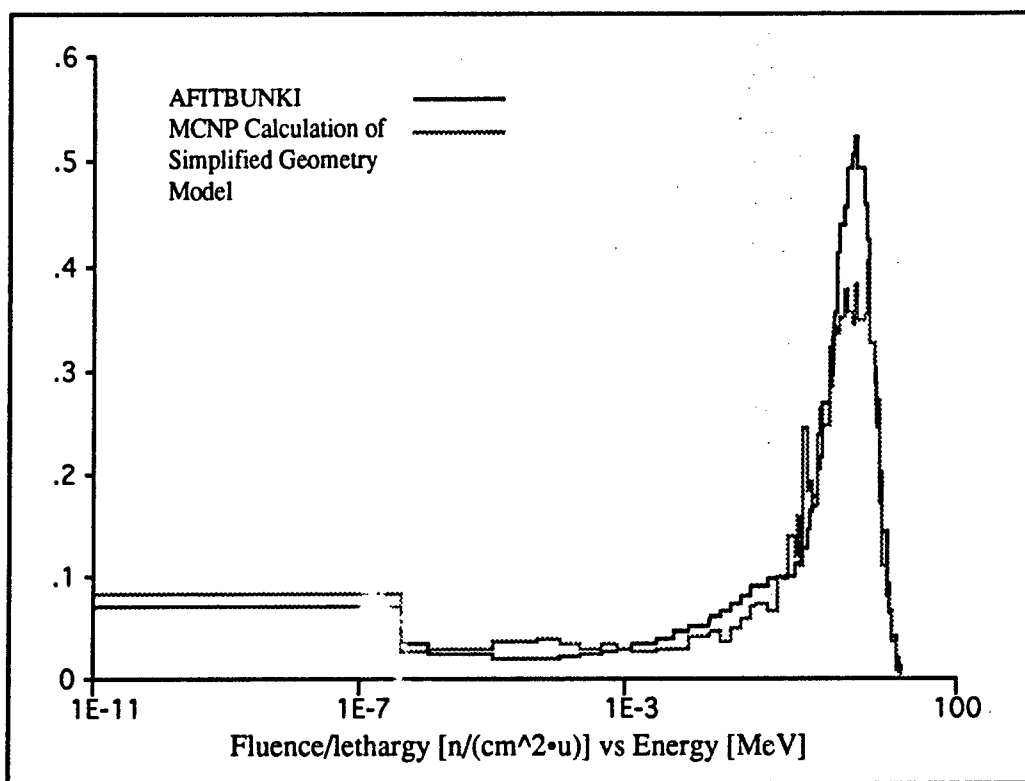


Figure 19. Polyethylene-moderated ²⁵²Cf Spectrum represented by a simplified geometry MCNP calculation and unfolded by AFITBUNKI using a MAXIET starting spectrum. The specified temperature in calculating the MAXIET spectrum was 1.1 MeV. The spectra are normalized to 1 source neutron/cm².

Table 12 summarizes the results of the validation of AFITBUNKI. The minimum and maximum percent differences between the calculated reference quantities and the AFITBUNKI-measured quantities are tabulated. The table also lists which starting spectrum caused the difference and whether AFITBUNKI over- or under-estimated the measurement. In the case of the two moderated ^{252}Cf spectra, AFITBUNKI always provided quantities which were higher than the reference values regardless of the starting

Table 12

SUMMARIZED COMPARISON OF AFITBUNKI
MEASUREMENTS AND CALCULATED REFERENCE VALUES

Minimum % Difference and Starting Spectrum			
Quantity Compared	Bare Cf-252	D20-Mod Cf-252	Poly-Mod Cf-252
\bar{E}	(-)1%, bare	7%, MAXIET	17%, MAXIET
H/Φ	(-)1%, flat	2%, MAXIET	14%, MAXIET
H^*/Φ (ICRP 26)	0%, flat	1%, MAXIET	14%, MAXIET
H^*/Φ (ICRP60)	(-)1%, flat	--	12%, MAXIET
Maximum % Difference and Starting Spectrum			
Quantity Compared	Bare Cf-252	D20-Mod Cf-252	Poly-Mod Cf-252
\bar{E}	5%, flat	39%, flat	33%, flat
H/Φ	(-)2%, bare	2%, flat	17%, flat
H^*/Φ (ICRP 26)	(-)1%, bare	5%, flat	17%, flat
H^*/Φ (ICRP60)	(-)2%, bare	--	15%, flat

All percent differences shown are relative to the reference values. AFITBUNKI measurements which are lower than the reference values are indicated by (-). The starting spectra used to unfold the measurements follow the percent difference; "bare" refers to a bare ^{252}Cf starting spectra.

spectrum used. This is acceptable because it provides dosimetric quantities which err on the conservative side. However, there also is a trend of smaller errors when more *a priori* information is applied to finding a solution as in the case when using the MAXIET starting spectrum. A MAXIET spectrum, with its small thermal peak, $1/E$ intermediate component, and Maxwellian high energy peak, more accurately approximates the moderated spectra and biases the unfolded spectra to a better fit. In the case of the bare ^{252}Cf spectra, AFITBUNKI provided dosimetric quantities lower than the reference; however, they never differed by more than a few percent.

VII. Conclusions

The stated objective of modifying BUNKI was to eliminate options in the code which have been proven less effective, refine the binning structure to produce more resolved spectra in a range of interest (10^{-11} to 14.92 MeV), and to extend the thermal bin to include lower energies (10^{-11} MeV versus 10^{-8} MeV). Additional objectives were to include in the code an assortment of initial-guess spectra and a number of newly-defined dosimetric conversion factors.

Without a doubt, AFITBUNKI has fewer options of response matrices and unfolding algorithms. UTA54, the collapsed 171 energy-group response matrix, and the SPUNIT unfolding algorithm are the only options when using AFITBUNKI to unfold spectra.

The binning structure has been refined in AFITBUNKI so that there are twice as many energy groups over the range of interest as there are in BUNKI and BUNKIUT. The plots of AFITBUNKI- and BUNKIUT-unfolded spectra in Section IV clearly show the finer binning structure of AFITBUNKI at high energies. The plots of the unfolded polyethylene-moderated ^{252}Cf spectra (Figures 9 and 10) demonstrate that the most thermal response of the UTA4 response matrix differ from those of AFITBUNKI's UTA54 response matrix. A comparison of AFITBUNKI-unfolded polyethylene-moderated ^{252}Cf spectra with the MCNP-calculated spectra (Figures 18 and 19) shows the UTA54 response to be relatively accurate in the most thermal bin.

Seven different starting spectra have been included in AFITBUNKI and results from three have been presented. Based on a comparison of the AFITBUNKI-unfolded bare ^{252}Cf spectra and the NIST fit (Figure 15), using the "answer" as an initial guess seems to provide the best results. While starting with a flat spectrum provides reasonable approximations of the spectrum, using the bare ^{252}Cf starting spectra provided a much more accurate result. In the case of the two moderated ^{252}Cf fission spectra, the MAXIET-calculated initial guess spectra provides the best starting point and AFITBUNKI yielded spectra that were much more reasonable in appearance than those calculated using a flat starting spectra. This is a good example of getting better results when more *a priori* knowledge is applied to find a solution.

With sufficient information applied to the solution via the initial guess spectrum, the spectra unfolded by AFITBUNKI display reasonable accuracy and the energy-integrated quantities agree with reference values even better. Energy-integrated dosimetric quantities calculated by AFITBUNKI tend to range between being within a few percent to being more conservative than the references. AFITBUNKI's tendency to over-estimate dosimetric quantities is the preferable alternative from the standpoint of safety.

Given the proper computational facilities, MCNP can be used to model complex geometries and perform numerous transport calculations on them. Hertel and Davidson's 171 energy-group response matrix is seemingly very effective. However, one way to improve the spectral results of AFITBUNKI might be to use MCNP to more accurately model the geometry of the Bonner sphere detector system. Using the most recent cross-section data and the exact energy binning structure, a more accurate response matrix could be calculated. In any case, it must be kept in mind that the matrix approximation of

the Fredholm equation will always be underdetermined when there are more energy intervals than detectors and a unique solution cannot exist. It could very well be that unfolding with a more accurate response matrix will not provide more accurate results without applying better or more *a priori* information via the initial guess spectrum.

The attempt to bound the error in AFITBUNKI measurements has been limited to stating the variation in energy-integrated quantities resulting from varying the initial guess spectrum. Performing a complete error analysis would require controlling so many variables that this would be a challenging task. Variables to be considered are the starting spectrum used, error in the Bonner sphere detector data, number of iterations between end-tests, and the specified end test.

AFITBUNKI is an effective code for unfolding Bonner sphere detector data. By using an initial-guess spectrum with a shape which is similar to the shape of the observed spectrum, AFITBUNKI unfolds the observed spectrum with subjectively reasonable accuracy and calculates energy-integrated dosimetric quantities which range from being within a few percent to as much as 20% erring on the conservative side.

VIII. Summary

The neutron spectrum unfolding code BUNKI, developed at the Naval Research Laboratory in 1983, has been modified to incorporate finer energy group structuring and updated to include more recent dosimetric conversion factors. The modified code, AFITBUNKI, unfolds spectra into 54 energy groups between 10^{-11} and 14.92 MeV and calculates fluence, absorbed dose, percent of effective dose equivalent, and percent of ambient dose equivalent as a function of neutron energy. It also calculates total energy-integrated fluence, absorbed dose (D), dose equivalent (H), ambient dose (D^*), both ICRP Publication 26- and ICRP Publication 60-based ambient dose equivalent (H^*), effective dose equivalent (H_E), and effective dose (E), as well as the effective quality factor (\overline{Q}), and average neutron energy. AFITBUNKI incorporates BUNKI's SPUNIT iterative unfolding algorithm and UTA54, a 171-energy group response matrix appropriately collapsed.

The user specifies an initial spectrum or directs MAXIET, a subroutine also derived from BUNKI, to calculate a $(1/E)$ plus a Maxwellian spectrum as an initial guess. The starting spectra than can be specified from within AFITBUNKI are flat, bare ^{252}Cf fission spectrum, D_2O -moderated ^{252}Cf fission spectrum, ^{252}Cf room return spectrum, D-T fusion spectrum, Am-Be fission spectrum, and D-Be fusion spectrum.

The code has been verified by a acceptable comparison of spectra unfolded by AFITBUNKI with spectra unfolded by BUNKIUT. AFITBUNKI has been validated against calculations of a NIST analytical fit of ^{252}Cf neutrons and against calculations of

both D₂O- and polyethylene-moderated ²⁵²Cf fission neutrons made by Los Alamos National Laboratory's Monte Carlo Neutron-Photon transport code, MCNP. The bare ²⁵²Cf and D₂O-moderated ²⁵²Cf spectra were chosen because they are well described and recommended for use as calibration references. The polyethylene-moderated ²⁵²Cf spectra was chosen because it has a significant number of neutrons in the thermal region.

Appendix A: AFITBUNKI Code

PROGRAM AFITBUNKI - AIR FORCE INSTITUTE OF TECHNOLOGY/ENP

25 FEBRUARY 1993

THIS PROGRAM IS A MODIFICATION OF THE NEUTRON UNFOLDING CODE BUNKI WHICH WAS DEVELOPED AT THE NAVAL RESEARCH LAB IN JULY 1983 BY KIMBERLY A LOWRY AND TOMMY L. JOHNSON. THIS PROGRAM CALCULATES NEUTRON FLUENCE, ABSORBED DOSE, PERCENT OF EFFECTIVE DOSE EQUIVALENT, AND PERCENT OF AMBIENT DOSE EQUIVALENT AS A FUNCTION OF ENERGY OVER 54 ENERGY INTERVALS BETWEEN 1E-11 AND 14.92 MeV. IN ADDITION TO THESE SPECTRAL QUANTITIES, THE CODE CALCULATES TOTAL ENERGY INTEGRATED FLUENCE, ABSORBED DOSE, DOSE EQUIVALENT, EFFECTIVE QUALITY FACTOR, AMBIENT DOSE, BOTH ICRP 26- AND ICRP 60-BASED AMBIENT DOSE EQUIVALENT AND EFFECTIVE AMBIENT QUALITY FACTOR, EFFECTIVE DOSE EQUIVALENT, EFFECTIVE DOSE, AND AVERAGE NEUTRON ENERGY. AFITBUNKI USES THE SPUNIT UNFOLDING ALGORITHM AND UTA54, THE 54 ENERGY GROUP RESPONSE MATRIX WHICH HAS BEEN COLLAPSED FROM N.E. HERTEL AND J.W. DAVIDSON'S 171 ENERGY GROUP RESPONSE MATRIX FOR 4mm x 4mm LiI BONNER SPHERE DETECTORS. THE USER MAY SPECIFY ONE OF THE SEVEN INITIAL SPECTRUMS INCORPORATED IN AFITBUNKI, OR MAY CALCULATE A 1/E PLUS A MAXWELLIAN SPECTRUM USING THE MAXIET ALGORITHM, OR MAY READ IN A SPECTRUM. OUTPUT FILES FROM THE CODE INCLUDE A TWO-PAGE SUMMARY DATA FILE AND A 54 ELEMENT COMMA-SEPARATED LIST OF THE FLUENCE/LETHARGY.

THIS CODE WAS DEVELOPED BY SEAN C. MILLER AT THE AIR FORCE INSTITUTE OF TECHNOLOGY AS PART OF HIS MASTER'S THESIS PROGRAM

CHARACTER*1 INSPFLAG,HAVEFITFLAG
CHARACTER*1 HAVEDETSFLAG,HAVEMAXFLAG
CHARACTER*4 CHDET,CODE
CHARACTER*4 CHISPC,CHTEMP,CHFIT,CHBCE
CHARACTER*8 ball,BALLANS, LISTNM
CHARACTER*21 INSPNAM
CHARACTER*10 FILNAMH
CHARACTER*63 BALFILE, SPFILE, TKFILENM, FILNAM, HEAD
CHARACTER*4 SAVE, LASTSP, CHMTX
DIMENSION ALETH(13,54),SPC(54),BCE(13),BCC(13),
& CRAD(54),CREM(54),CEFF(54),CAMD(54),CAMH2(54),CAMH6(54),
& CHEFF(54),CHEFFR(54),CEFFI(54),
& RAD(54),REM(54),EEND(55),CE(54),SPL(54),WDLETH(54),

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& SPLI(54), SPL1(54), SPL3(54),
& SPL4(54), SPL5(54), SPL6(54), SPL7(54), SPL11(54),
& SPLM(55), SPLMAX(55), LL(13), BALL(13), PCTERR(13), CODE(13),
& PHEFF(54), PAMH6(54), ERRBCE(13), WHTBCE(13), A(13,54),
& SPLL(54), SS(54)
C
C JJ IS THE NUM OF ENERGY INTERVALS AND JJJ THE NUM OF ENERGY ENDPOINTS
  INTEGER JJ/54/, JJJ/55/
C
C ENERGY INTERVAL END POINTS - lowest to highest (JJJ values)
  DATA EEND/
    & 1.000E-11, 4.140E-07, 1.125E-06, 1.068E-05, 4.785E-05, 1.013E-04,
    & 2.145E-04, 4.540E-04, 7.485E-04, 1.234E-03, 3.035E-03, 5.531E-03,
    & 9.119E-03, 1.931E-02, 2.850E-02, 4.087E-02, 5.656E-02, 7.950E-02,
    & 9.804E-02, 1.500E-01, 2.024E-01, 2.472E-01, 2.985E-01, 3.688E-01,
    & 4.076E-01, 4.505E-01, 4.979E-01, 5.502E-01, 6.081E-01, 7.065E-01,
    & 8.209E-01, 9.072E-01, 1.003E+00, 1.225E+00, 1.423E+00, 1.572E+00,
    & 1.827E+00, 2.019E+00, 2.307E+00, 2.592E+00, 3.012E+00, 3.329E+00,
    & 4.066E+00, 4.493E+00, 4.966E+00, 6.055E+00, 7.047E+00, 7.788E+00,
    & 9.048E+00, 1.000E+01, 1.105E+01, 1.221E+01, 1.284E+01, 1.419E+01,
    & 1.492E+01/
C
C NAMES OF THE DETECTORS
  DATA BALL/
    & 'bare', 'bare+cd', '2 inch', '2"+cd',
    & '3 inch', '3"+cd', '5 inch', '5"+cd',
    & '8 inch', '10 inch', '12 inch', '15 inch', '18 inch' /
C
C CONVERSION FACTORS ALL START WITH THE VALUE FOR THE LOWEST ENERGY BIN
C
C FLUENCE TO DOSE CONVERSION FACTORS (ICRP 21) - D(E)/F(E)
C (pGy-cm^2/neutron) - JJ values - average CF over interval
  DATA CRAD/
    & 5.371E+00, 6.261E+00, 6.171E+00, 5.963E+00, 5.838E+00, 5.675E+00,
    & 5.458E+00, 5.281E+00, 5.146E+00, 4.977E+00, 4.867E+00, 4.895E+00,
    & 5.150E+00, 5.552E+00, 5.946E+00, 6.403E+00, 6.964E+00, 7.528E+00,
    & 8.478E+00, 9.928E+00, 1.122E+01, 1.245E+01, 1.395E+01, 1.531E+01,
    & 1.631E+01, 1.740E+01, 1.861E+01, 1.997E+01, 2.193E+01, 2.470E+01,
    & 2.737E+01, 2.968E+01, 3.274E+01, 3.601E+01, 3.820E+01, 4.019E+01,
    & 4.207E+01, 4.358E+01, 4.504E+01, 4.651E+01, 4.784E+01, 4.935E+01,
    & 5.085E+01, 5.182E+01, 5.335E+01, 5.521E+01, 5.657E+01, 5.798E+01,
    & 5.946E+01, 6.076E+01, 6.218E+01, 6.329E+01, 6.446E+01, 6.566E+01/
C
C FLUENCE TO DOSE EQUIVALENT CONVERSION FACTORS (ICRP 21) - H(E)/F(E)
C (pSv-cm^2/neutron) - Cubic interpolation of Log-Log values from
C ICRP 21, TABLE 4 and FIG 14 - JJ values - average CF over interval

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DATA CREM/

& 1.148E+01,	1.248E+01,	1.234E+01,	1.193E+01,	1.168E+01,	1.135E+01,
& 1.092E+01,	1.056E+01,	1.019E+01,	9.467E+00,	9.017E+00,	9.364E+00,
& 1.199E+01,	1.709E+01,	2.279E+01,	3.025E+01,	4.041E+01,	5.169E+01,
& 6.702E+01,	8.865E+01,	1.074E+02,	1.246E+02,	1.452E+02,	1.636E+02,
& 1.765E+02,	1.904E+02,	2.058E+02,	2.227E+02,	2.455E+02,	2.747E+02,
& 2.996E+02,	3.183E+02,	3.414E+02,	3.644E+02,	3.773E+02,	3.869E+02,
& 3.947E+02,	3.994E+02,	4.031E+02,	4.058E+02,	4.075E+02,	4.084E+02,
& 4.088E+02,	4.087E+02,	4.085E+02,	4.083E+02,	4.080E+02,	4.080E+02,
& 4.083E+02,	4.094E+02,	4.115E+02,	4.133E+02,	4.153E+02,	4.174E+02/

C
C FLUENCE TO EFFECTIVE DOSE (AP EXPOSURE) CONVERSION FACTORS - E(E)/F(E)
C (pSv-cm²/neutron) - ICRP 60 BASED, Cubic interpolation of Log-Log
C values Jennifer Tanner, PNL, Research Data - JJ values - average CF
C over interval

DATA CEFF/

& 1.431E+01,	1.410E+01,	1.412E+01,	1.417E+01,	1.423E+01,	1.429E+01,
& 1.436E+01,	1.444E+01,	1.453E+01,	1.476E+01,	1.514E+01,	1.566E+01,
& 1.935E+01,	2.613E+01,	3.236E+01,	3.963E+01,	4.847E+01,	5.750E+01,
& 7.093E+01,	9.081E+01,	1.083E+02,	1.245E+02,	1.433E+02,	1.596E+02,
& 1.706E+02,	1.820E+02,	1.927E+02,	2.021E+02,	2.136E+02,	2.275E+02,
& 2.394E+02,	2.497E+02,	2.712E+02,	3.021E+02,	3.257E+02,	3.530E+02,
& 3.804E+02,	4.011E+02,	4.175E+02,	4.302E+02,	4.436E+02,	4.634E+02,
& 4.762E+02,	4.744E+02,	4.750E+02,	4.786E+02,	4.803E+02,	4.812E+02,
& 4.813E+02,	4.812E+02,	4.813E+02,	4.813E+02,	4.812E+02,	4.812E+02/

C
C FLUENCE TO EFFECTIVE DOSE (ISOTROPIC EXP) CONV FACTORS - E(E)/F(E)
C (pSv-cm²/neutron) - ICRP 60 BASED, Cubic interpolation of Log-Log
C values Jennifer Tanner, PNL, Research Data - JJ values - average CF
C over interval

DATA CEFFI/

& 6.665E+00,	6.581E+00,	6.589E+00,	6.610E+00,	6.634E+00,	6.658E+00,
& 6.687E+00,	6.718E+00,	6.752E+00,	6.837E+00,	6.976E+00,	7.168E+00,
& 8.933E+00,	1.220E+01,	1.501E+01,	1.803E+01,	2.127E+01,	2.432E+01,
& 2.943E+01,	3.747E+01,	4.467E+01,	5.125E+01,	5.891E+01,	6.554E+01,
& 7.006E+01,	7.477E+01,	7.914E+01,	8.299E+01,	8.779E+01,	9.384E+01,
& 9.937E+01,	1.045E+02,	1.160E+02,	1.335E+02,	1.482E+02,	1.648E+02,
& 1.829E+02,	1.981E+02,	2.120E+02,	2.243E+02,	2.348E+02,	2.469E+02,
& 2.607E+02,	2.722E+02,	2.826E+02,	2.919E+02,	2.995E+02,	3.053E+02,
& 3.085E+02,	3.093E+02,	3.096E+02,	3.097E+02,	3.095E+02,	3.092E+02/

C
C FLUENCE TO EFFECTIVE DOSE EQUIVALENT CONVERSION FACTORS, AP EXPOSURE
C - HE(E)/F(E) - (pSv-cm²/neutron) - ICRP 26/51 BASED - PRE-1985
C QUALITY FACTORS, Cubic interpolation of Log-Log values from ICRP 51,
C pg 32.
C

DATA CHEFF/

& 4.343E+00, 4.764E+00, 4.628E+00, 4.344E+00, 4.188E+00, 4.061E+00,
 & 3.930E+00, 3.855E+00, 3.845E+00, 3.946E+00, 4.137E+00, 4.355E+00,
 & 5.056E+00, 6.460E+00, 8.278E+00, 1.061E+01, 1.388E+01, 1.766E+01,
 & 2.385E+01, 3.380E+01, 4.279E+01, 5.109E+01, 6.109E+01, 7.006E+01,
 & 7.636E+01, 8.312E+01, 9.015E+01, 9.736E+01, 1.068E+02, 1.190E+02,
 & 1.297E+02, 1.387E+02, 1.527E+02, 1.701E+02, 1.828E+02, 1.959E+02,
 & 2.095E+02, 2.228E+02, 2.379E+02, 2.547E+02, 2.708E+02, 2.893E+02,
 & 3.081E+02, 3.202E+02, 3.370E+02, 3.569E+02, 3.712E+02, 3.858E+02,
 & 4.025E+02, 4.196E+02, 4.398E+02, 4.558E+02, 4.717E+02, 4.859E+02/

C

C FLUENCE TO EFFECTIVE DOSE EQUIVALENT CONVERSION FACTORS, ROT EXPOSURE

C - HE(E)/F(E) - (pSv-cm²/neutron) - ICRP 26/51 BASED - PRE-1985

C QUALITY FACTORS, Cubic interpolation of Log-Log values from ICRP 51,

C pg 32.

DATA CHEFFR/

& 2.427E+00, 2.595E+00, 2.551E+00, 2.426E+00, 2.353E+00, 2.295E+00,
 & 2.234E+00, 2.197E+00, 2.186E+00, 2.215E+00, 2.277E+00, 2.350E+00,
 & 2.602E+00, 3.108E+00, 3.766E+00, 4.613E+00, 5.847E+00, 7.312E+00,
 & 9.661E+00, 1.344E+01, 1.716E+01, 2.097E+01, 2.575E+01, 3.018E+01,
 & 3.335E+01, 3.680E+01, 4.029E+01, 4.373E+01, 4.822E+01, 5.397E+01,
 & 5.912E+01, 6.353E+01, 7.072E+01, 8.351E+01, 9.341E+01, 1.043E+02,
 & 1.160E+02, 1.272E+02, 1.373E+02, 1.540E+02, 1.681E+02, 1.850E+02,
 & 2.023E+02, 2.130E+02, 2.260E+02, 2.458E+02, 2.584E+02, 2.708E+02,
 & 2.852E+02, 3.016E+02, 3.222E+02, 3.389E+02, 3.560E+02, 3.714E+02/

C

C FLUENCE TO AMBIENT DOSE CONVERSION FACTORS - D*(E)/F(E)

C (pGY-cm²/neutron) - ICRP 26 BASED, RPD 12:231-235(1985) -WAGNER'S

C FERMI-LORENZ FIT EQN- JJ values - average CF over interval

DATA CAMD/

& 2.706E+00, 3.041E+00, 2.990E+00, 2.757E+00, 2.551E+00, 2.426E+00,
 & 2.330E+00, 2.279E+00, 2.268E+00, 2.319E+00, 2.454E+00, 2.654E+00,
 & 3.060E+00, 3.612E+00, 4.160E+00, 4.825E+00, 5.673E+00, 6.552E+00,
 & 7.851E+00, 9.734E+00, 1.135E+01, 1.281E+01, 1.452E+01, 1.600E+01,
 & 1.702E+01, 1.810E+01, 1.924E+01, 2.044E+01, 2.201E+01, 2.402E+01,
 & 2.579E+01, 2.727E+01, 2.957E+01, 3.240E+01, 3.450E+01, 3.664E+01,
 & 3.885E+01, 4.093E+01, 4.319E+01, 4.562E+01, 4.793E+01, 5.065E+01,
 & 5.341E+01, 5.522E+01, 5.786E+01, 6.093E+01, 6.306E+01, 6.512E+01,
 & 6.714E+01, 6.869E+01, 7.021E+01, 7.132E+01, 7.240E+01, 7.346E+01/

C

C FLUENCE TO AMBIENT DOSE EQUIVALENT CONVERSION FACTORS - H*(E)/F(E)

C (pSv-cm²/neutron) - ICRP 26 BASED, RPD 12:231-235(1985) - WAGNER'S

C FERMI-LORENZ FIT EQN- JJ values - average CF over interval

DATA CAMH2/

& 9.320E+00, 1.041E+01, 9.911E+00, 8.712E+00, 7.753E+00, 7.135E+00,

& 6.618E+00, 6.296E+00, 6.164E+00, 6.321E+00, 6.968E+00, 8.202E+00,
 & 1.137E+01, 1.683E+01, 2.346E+01, 3.284E+01, 4.632E+01, 6.170E+01,
 & 8.473E+01, 1.192E+02, 1.474E+02, 1.710E+02, 1.963E+02, 2.168E+02,
 & 2.297E+02, 2.426E+02, 2.552E+02, 2.676E+02, 2.825E+02, 2.997E+02,
 & 3.134E+02, 3.236E+02, 3.377E+02, 3.530E+02, 3.628E+02, 3.714E+02,
 & 3.795E+02, 3.860E+02, 3.923E+02, 3.981E+02, 4.029E+02, 4.075E+02,
 & 4.116E+02, 4.139E+02, 4.171E+02, 4.212E+02, 4.253E+02, 4.328E+02,
 & 4.446E+02, 4.601E+02, 4.819E+02, 5.020E+02, 5.248E+02, 5.482E+02/

C

C FLUENCE TO AMBIENT DOSE EQUIVALENT CONVERSION FACTORS - H*(E)/F(E)

C (pSv-cm²/neutron) - ICRP 60 BASED, RPD 40,2:85-89(1992) -

C SCHUHMACHER'S COEFFS TO WAGNERS FERMI-LORENZ FIT EQN -

C JJ values - average CF over interval

DATA CAMH6/

& 1.253E+01, 1.273E+01, 1.195E+01, 1.071E+01, 9.822E+00, 9.222E+00,
 & 8.677E+00, 8.297E+00, 8.092E+00, 8.134E+00, 8.699E+00, 1.002E+01,
 & 1.397E+01, 2.148E+01, 3.171E+01, 4.739E+01, 7.130E+01, 9.958E+01,
 & 1.413E+02, 2.023E+02, 2.489E+02, 2.850E+02, 3.208E+02, 3.478E+02,
 & 3.637E+02, 3.788E+02, 3.929E+02, 4.059E+02, 4.206E+02, 4.363E+02,
 & 4.479E+02, 4.558E+02, 4.657E+02, 4.753E+02, 4.806E+02, 4.846E+02,
 & 4.878E+02, 4.898E+02, 4.912E+02, 4.919E+02, 4.920E+02, 4.912E+02,
 & 4.899E+02, 4.887E+02, 4.863E+02, 4.832E+02, 4.807E+02, 4.781E+02,
 & 4.758E+02, 4.755E+02, 4.801E+02, 4.895E+02, 5.125E+02, 5.424E+02/

C

C UTA4 13-BALL, 54-ENERGY GROUP RESPONSE MATRIX

C I = DETECTOR NUMBER; J = ENERGY GROUP

C Every 9 lines of data correspond to one ball response, starting

C with the bare det and ending with the 18 inch. The following data

C is listed in order of highest to lowest energy. (The first data point

C is bare at 14.92 MeV) See the exact ball order at line 1130.

C

DATA ((A(J,I), I=54,1,-1), J=1,13) /

& 5.525E-05, 4.605E-05, 5.380E-05, 6.130E-05, 6.290E-05, 5.870E-05,
 & 7.567E-05, 8.385E-05, 7.604E-05, 9.387E-05, 8.950E-05, 9.770E-05,
 & 1.205E-04, 1.420E-04, 1.647E-04, 1.976E-04, 2.249E-04, 2.485E-04,
 & 2.603E-04, 2.555E-04, 2.583E-04, 2.518E-04, 2.529E-04, 2.615E-04,
 & 2.767E-04, 3.040E-04, 3.400E-04, 3.820E-04, 4.420E-04, 5.340E-04,
 & 6.785E-04, 1.174E-03, 2.578E-03, 2.875E-03, 1.362E-03, 7.611E-04,
 & 6.444E-04, 6.944E-04, 7.728E-04, 8.225E-04, 1.012E-03, 1.323E-03,
 & 1.780E-03, 2.336E-03, 3.399E-03, 4.765E-03, 6.090E-03, 8.290E-03,
 & 1.193E-02, 1.707E-02, 2.891E-02, 6.345E-02, 1.055E-01, 1.526E-01,
 & 6.480E-05, 5.540E-05, 6.230E-05, 6.835E-05, 6.765E-05, 6.190E-05,
 & 7.837E-05, 8.645E-05, 7.896E-05, 9.692E-05, 9.285E-05, 1.010E-04,
 & 1.235E-04, 1.455E-04, 1.687E-04, 2.021E-04, 2.306E-04, 2.560E-04,
 & 2.660E-04, 2.600E-04, 2.587E-04, 2.523E-04, 2.555E-04, 2.675E-04,
 & 2.783E-04, 3.057E-04, 3.420E-04, 3.840E-04, 4.430E-04, 5.340E-04,

& 6.790E-04, 1.176E-03, 2.581E-03, 2.857E-03, 1.360E-03, 7.602E-04,
 & 6.443E-04, 6.941E-04, 7.725E-04, 8.215E-04, 1.009E-03, 1.320E-03,
 & 1.780E-03, 2.326E-03, 3.382E-03, 4.745E-03, 6.050E-03, 8.137E-03,
 & 1.153E-02, 1.600E-02, 2.872E-02, 6.291E-02, 7.955E-02, 2.914E-06,
 & 4.735E-04, 4.735E-04, 5.060E-04, 5.435E-04, 5.910E-04, 6.485E-04,
 & 7.880E-04, 9.445E-04, 1.119E-03, 1.422E-03, 1.640E-03, 1.860E-03,
 & 2.255E-03, 2.690E-03, 3.107E-03, 3.620E-03, 4.145E-03, 4.695E-03,
 & 5.333E-03, 6.025E-03, 6.790E-03, 7.962E-03, 9.077E-03, 9.890E-03,
 & 1.097E-02, 1.240E-02, 1.365E-02, 1.465E-02, 1.570E-02, 1.690E-02,
 & 1.800E-02, 1.993E-02, 2.256E-02, 2.470E-02, 2.712E-02, 3.112E-02,
 & 3.467E-02, 3.765E-02, 4.125E-02, 4.488E-02, 4.890E-02, 5.513E-02,
 & 6.245E-02, 6.947E-02, 8.002E-02, 9.085E-02, 9.960E-02, 1.110E-01,
 & 1.260E-01, 1.423E-01, 1.690E-01, 2.102E-01, 2.270E-01, 1.324E-01,
 & 8.360E-04, 8.240E-04, 8.290E-04, 8.160E-04, 7.845E-04, 7.880E-04,
 & 9.173E-04, 1.075E-03, 1.266E-03, 1.585E-03, 1.815E-03, 2.040E-03,
 & 2.455E-03, 2.910E-03, 3.350E-03, 3.880E-03, 4.440E-03, 5.020E-03,
 & 5.723E-03, 6.410E-03, 7.067E-03, 8.270E-03, 9.419E-03, 1.025E-02,
 & 1.137E-02, 1.280E-02, 1.405E-02, 1.510E-02, 1.620E-02, 1.730E-02,
 & 1.850E-02, 2.048E-02, 2.311E-02, 2.527E-02, 2.772E-02, 3.172E-02,
 & 3.527E-02, 3.823E-02, 4.181E-02, 4.543E-02, 4.946E-02, 5.560E-02,
 & 6.290E-02, 6.982E-02, 8.028E-02, 8.985E-02, 9.845E-02, 1.067E-01,
 & 1.207E-01, 1.293E-01, 1.665E-01, 2.069E-01, 1.615E-01, 7.556E-07,
 & 4.170E-03, 4.322E-03, 4.590E-03, 4.905E-03, 5.345E-03, 5.895E-03,
 & 7.107E-03, 8.475E-03, 1.010E-02, 1.245E-02, 1.420E-02, 1.590E-02,
 & 1.865E-02, 2.180E-02, 2.463E-02, 2.808E-02, 3.141E-02, 3.485E-02,
 & 3.870E-02, 4.270E-02, 4.683E-02, 5.287E-02, 5.825E-02, 6.190E-02,
 & 6.653E-02, 7.207E-02, 7.670E-02, 8.030E-02, 8.380E-02, 8.730E-02,
 & 9.075E-02, 9.588E-02, 1.018E-01, 1.072E-01, 1.133E-01, 1.208E-01,
 & 1.268E-01, 1.309E-01, 1.363E-01, 1.413E-01, 1.463E-01, 1.543E-01,
 & 1.630E-01, 1.714E-01, 1.821E-01, 1.930E-01, 2.005E-01, 2.097E-01,
 & 2.203E-01, 2.300E-01, 2.413E-01, 2.446E-01, 2.223E-01, 1.055E-01,
 & 5.650E-03, 5.768E-03, 5.940E-03, 6.075E-03, 6.240E-03, 6.600E-03,
 & 7.787E-03, 9.170E-03, 1.081E-02, 1.325E-02, 1.505E-02, 1.680E-02,
 & 1.955E-02, 2.270E-02, 2.563E-02, 2.910E-02, 3.249E-02, 3.600E-02,
 & 3.990E-02, 4.380E-02, 4.773E-02, 5.382E-02, 5.921E-02, 6.285E-02,
 & 6.747E-02, 7.300E-02, 7.760E-02, 8.120E-02, 8.470E-02, 8.820E-02,
 & 9.165E-02, 9.663E-02, 1.028E-01, 1.080E-01, 1.138E-01, 1.215E-01,
 & 1.268E-01, 1.314E-01, 1.363E-01, 1.411E-01, 1.462E-01, 1.537E-01,
 & 1.625E-01, 1.706E-01, 1.815E-01, 1.890E-01, 1.975E-01, 2.003E-01,
 & 2.100E-01, 2.077E-01, 2.375E-01, 2.405E-01, 1.582E-01, 5.181E-07,
 & 3.095E-02, 3.233E-02, 3.420E-02, 3.650E-02, 3.960E-02, 4.285E-02,
 & 5.070E-02, 5.870E-02, 6.778E-02, 7.952E-02, 8.835E-02, 9.560E-02,
 & 1.055E-01, 1.185E-01, 1.287E-01, 1.416E-01, 1.518E-01, 1.625E-01,
 & 1.730E-01, 1.820E-01, 1.910E-01, 2.022E-01, 2.102E-01, 2.150E-01,
 & 2.203E-01, 2.257E-01, 2.295E-01, 2.315E-01, 2.330E-01, 2.340E-01,
 & 2.350E-01, 2.350E-01, 2.345E-01, 2.325E-01, 2.293E-01, 2.241E-01,

& 2.194E-01, 2.158E-01, 2.118E-01, 2.085E-01, 2.060E-01, 2.030E-01,
 & 2.015E-01, 2.000E-01, 1.984E-01, 1.965E-01, 1.945E-01, 1.920E-01,
 & 1.883E-01, 1.837E-01, 1.742E-01, 1.517E-01, 1.238E-01, 5.385E-02,
 & 3.420E-02, 3.548E-02, 3.720E-02, 3.925E-02, 4.180E-02, 4.470E-02,
 & 5.247E-02, 6.050E-02, 6.948E-02, 8.120E-02, 9.000E-02, 9.720E-02,
 & 1.070E-01, 1.200E-01, 1.300E-01, 1.430E-01, 1.530E-01, 1.640E-01,
 & 1.737E-01, 1.830E-01, 1.917E-01, 2.022E-01, 2.107E-01, 2.150E-01,
 & 2.207E-01, 2.257E-01, 2.285E-01, 2.310E-01, 2.320E-01, 2.340E-01,
 & 2.340E-01, 2.340E-01, 2.335E-01, 2.315E-01, 2.282E-01, 2.227E-01,
 & 2.176E-01, 2.141E-01, 2.098E-01, 2.072E-01, 2.044E-01, 2.020E-01,
 & 1.995E-01, 1.984E-01, 1.964E-01, 1.920E-01, 1.905E-01, 1.823E-01,
 & 1.787E-01, 1.650E-01, 1.708E-01, 1.490E-01, 8.725E-02, 2.553E-07,
 & 8.705E-02, 9.075E-02, 9.560E-02, 1.017E-01, 1.085E-01, 1.140E-01,
 & 1.307E-01, 1.450E-01, 1.611E-01, 1.780E-01, 1.900E-01, 1.950E-01,
 & 1.975E-01, 2.125E-01, 2.190E-01, 2.319E-01, 2.352E-01, 2.405E-01,
 & 2.410E-01, 2.405E-01, 2.377E-01, 2.315E-01, 2.248E-01, 2.195E-01,
 & 2.120E-01, 2.027E-01, 1.945E-01, 1.885E-01, 1.820E-01, 1.750E-01,
 & 1.685E-01, 1.597E-01, 1.484E-01, 1.383E-01, 1.270E-01, 1.144E-01,
 & 1.050E-01, 9.945E-02, 9.364E-02, 8.899E-02, 8.502E-02, 8.050E-02,
 & 7.660E-02, 7.375E-02, 7.033E-02, 6.735E-02, 6.530E-02, 6.273E-02,
 & 5.970E-02, 5.660E-02, 5.191E-02, 4.348E-02, 3.465E-02, 1.512E-02,
 & 1.110E-01, 1.155E-01, 1.210E-01, 1.280E-01, 1.355E-01, 1.390E-01,
 & 1.563E-01, 1.690E-01, 1.829E-01, 1.945E-01, 2.020E-01, 2.000E-01,
 & 1.915E-01, 2.010E-01, 1.997E-01, 2.064E-01, 2.012E-01, 2.000E-01,
 & 1.930E-01, 1.850E-01, 1.760E-01, 1.620E-01, 1.497E-01, 1.420E-01,
 & 1.320E-01, 1.210E-01, 1.120E-01, 1.055E-01, 9.910E-02, 9.330E-02,
 & 8.780E-02, 8.024E-02, 7.178E-02, 6.498E-02, 5.793E-02, 5.034E-02,
 & 4.543E-02, 4.230E-02, 3.934E-02, 3.704E-02, 3.509E-02, 3.293E-02,
 & 3.110E-02, 2.973E-02, 2.817E-02, 2.685E-02, 2.595E-02, 2.483E-02,
 & 2.353E-02, 2.227E-02, 2.035E-02, 1.700E-02, 1.358E-02, 5.942E-03,
 & 1.220E-01, 1.260E-01, 1.310E-01, 1.380E-01, 1.445E-01, 1.450E-01,
 & 1.593E-01, 1.680E-01, 1.782E-01, 1.825E-01, 1.830E-01, 1.750E-01,
 & 1.585E-01, 1.620E-01, 1.553E-01, 1.564E-01, 1.464E-01, 1.410E-01,
 & 1.307E-01, 1.200E-01, 1.097E-01, 9.530E-02, 8.400E-02, 7.695E-02,
 & 6.897E-02, 6.027E-02, 5.400E-02, 4.945E-02, 4.550E-02, 4.190E-02,
 & 3.865E-02, 3.436E-02, 2.985E-02, 2.643E-02, 2.310E-02, 1.969E-02,
 & 1.756E-02, 1.626E-02, 1.508E-02, 1.415E-02, 1.336E-02, 1.250E-02,
 & 1.175E-02, 1.123E-02, 1.064E-02, 1.010E-02, 9.765E-03, 9.343E-03,
 & 8.850E-03, 8.370E-03, 7.646E-03, 6.384E-03, 5.093E-03, 2.243E-03,
 & 1.215E-01, 1.242E-01, 1.280E-01, 1.325E-01, 1.360E-01, 1.320E-01,
 & 1.410E-01, 1.435E-01, 1.468E-01, 1.415E-01, 1.345E-01, 1.210E-01,
 & 1.006E-01, 9.870E-02, 8.963E-02, 8.655E-02, 7.619E-02, 6.965E-02,
 & 6.053E-02, 5.200E-02, 4.443E-02, 3.535E-02, 2.883E-02, 2.515E-02,
 & 2.127E-02, 1.747E-02, 1.495E-02, 1.320E-02, 1.180E-02, 1.060E-02,
 & 9.535E-03, 8.244E-03, 6.966E-03, 6.060E-03, 5.218E-03, 4.399E-03,
 & 3.907E-03, 3.610E-03, 3.335E-03, 3.129E-03, 2.952E-03, 2.763E-03,

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& 2.600E-03, 2.483E-03, 2.344E-03, 2.230E-03, 2.150E-03, 2.060E-03,
& 1.953E-03, 1.843E-03, 1.685E-03, 1.408E-03, 1.125E-03, 4.967E-04,
& 1.095E-01, 1.110E-01, 1.120E-01, 1.150E-01, 1.160E-01, 1.080E-01,
& 1.120E-01, 1.110E-01, 1.086E-01, 9.875E-02, 8.805E-02, 7.420E-02,
& 5.675E-02, 5.360E-02, 4.580E-02, 4.231E-02, 3.481E-02, 3.010E-02,
& 2.443E-02, 1.950E-02, 1.557E-02, 1.125E-02, 8.475E-03, 7.040E-03,
& 5.640E-03, 4.363E-03, 3.575E-03, 3.080E-03, 2.690E-03, 2.370E-03,
& 2.100E-03, 1.790E-03, 1.485E-03, 1.285E-03, 1.100E-03, 9.261E-04,
& 8.228E-04, 7.594E-04, 7.022E-04, 6.585E-04, 6.213E-04, 5.810E-04,
& 5.470E-04, 5.224E-04, 4.937E-04, 4.700E-04, 4.530E-04, 4.340E-04,
& 4.107E-04, 3.887E-04, 3.550E-04, 2.966E-04, 2.370E-04, 1.050E-04
& /

C
C COMPUTE AVERAGE ENERGY, AND LOG WIDTH OF ENERGY INTERVALS
DO 1000 I=1,54
    CE(I)=(EEND(I)*EEND(I+1))**.5
    WDLETH(I)=ALOG10(EEND(I+1))-ALOG10(EEND(I))
1000 CONTINUE
C INPUT FIXED INITIAL CONDITIONS AND DATA
    INSPFLAG='N'      !Is there an initial spectrum?
    HAVEFITFLAG='N'    !Are the fit parameters chosen?
    HAVEMAXFLAG='N'    !Look for a Max, 1/E spectrum?
    HAVEDETSFLAG='N'   !Are there any detectors selected yet?
    HAVEBALLDATA='N'   !Has any ball data been read in yet?
    CHDET='Y'          !Change the det selection? (Ask for detsets?)

C
C
C SLOPEJ=0 ! (0) INITIAL VALUE OF THE SLOPE OF THE 1/E
C PART OF MAXIET
C PERSLP=.01 ! (0.005 - 0.02) THE AMT BY WHICH THE SLOPE OF 1/E
C PART OF THE MAXIET SPECTRUM IS CHANGED IN
C SEARCHING FOR A BETTER FIT TO THE DATA
C THERMJ=1.0 ! (1.0) INIT VAL OF THE THERML BIN OF THE MAXIET SPT
C THMMIN=0.1 ! MIN ALLOWBL VALUE OF THE THRM BIN OF MAXIET SPT
C THMMAX=10.0 ! MAX ALLOWBL VALUE OF THE THRM BIN OF MAXIET SPT
C THMMIN AND THMMAX ARE SET FROM PHYSICAL
C CHARACTERISTICS OF THE RADIATION ENVIRONMENT
C DEAD=0.0 ! DEAD TIME OF THE INSTRUMENT USED TO DETERMINE
C THE DETECTOR COUNTS
C SHP=.01 ! THE MIN VALUE OF THE (I+1) BIN RELATIVE TO THE I BIN
C FOR THE INIT MAXIET SPT. USED TO LIMIT THE HI ENERGY
C ROLL OFF OF THE CALC MAXIET SPECTRUM
C TSTRAT=.999 ! (0.9, 0.99, 0.999, 1.1) MAX ALLOWABLE VAL OF THE
C ERROR ON THE FIT RELTV TO THE VALUE WHEN THE
C ERR WAS LAST TESTED-WILL NOT TERMINATE FIT IF >1.0
MM=JJ

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        SPMX=1.0
        INSPANS = 11
C      INPUT NUMBER AND KIND OF DETECTORS
1030    IF(HAVEDETSFLAG.EQ.'N')GO TO 1060
        WRITE(*,1040)
1040    FORMAT(' CHANGE DETECTORS?',/)
1050    FORMAT(A1)
        READ(*,1050)CHDET.
        IF(CHDET.EQ.'Y')GO TO 1060
        IF(CHDET.EQ.'N')GO TO 1195
        GO TO 1030
1060    WRITE(*,1070)
1070    FORMAT(/,' NUMBER OF DETECTORS?',/)
        READ(*,*)KK
1080    WRITE(*,1090)
1090    FORMAT(/,' TYPE DETECTOR CODES',/,/,
& ' ? FOR HELP', /,
& ' 6 FOR THE 2-,3-,5-,8-,10- & 12-INCH DETECTORS',/,
& ' 6B FOR THE BARE, 2-,3-,5-,8-,10- & 12-INCH DETECTORS',/,
& ' 7 FOR THE 2-,3-,5-,8-,10-,12- & 18-INCH DETECTORS) OR ',/,
& ' 7B FOR THE BARE, 2-,3-,5-,8-,10-,12- & 18-INCH DETECTORS)',/)
        READ(*,1100)CODE(1)
1100    FORMAT(A4)
        IF(CODE(1).EQ.'?')GO TO 1120
C      A RESPONSE OF 6, 6B, 7 OR 7B GETS THE USUAL 6 OR 7 DETECTORS
        IF(CODE(1).EQ.'6') THEN
            CODE(1) = '2'
            CODE(2) = '3'
            CODE(3) = '5'
            CODE(4) = '8'
            CODE(5) = '10'
            CODE(6) = '12'
        ELSE IF(CODE(1).EQ.'6B') THEN
            CODE(1) = '0'
            CODE(2) = '2'
            CODE(3) = '3'
            CODE(4) = '5'
            CODE(5) = '8'
            CODE(6) = '10'
            CODE(7) = '12'
        ELSE IF(CODE(1).EQ.'7') THEN
            CODE(1) = '2'
            CODE(2) = '3'
            CODE(3) = '5'
            CODE(4) = '8'
            CODE(5) = '10'

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        CODE(6) = '12'
        CODE(7) = '18'
    ELSE IF (CODE(1).EQ.'7B') THEN
        CODE(1) = '0'
        CODE(2) = '2'
        CODE(3) = '3'
        CODE(4) = '5'
        CODE(5) = '8'
        CODE(6) = '10'
        CODE(7) = '12'
CODE(8) = '18'
    ELSE
        READ(*,1100) (CODE(I),I=2,KK)
    END IF
    DO 1110 I=1,KK
        LL(I)=0
        IF (CODE(I).EQ.'0') LL(I)=1
        IF (CODE(I).EQ.'0C') LL(I)=2
        IF (CODE(I).EQ.'2') LL(I)=3
        IF (CODE(I).EQ.'2C') LL(I)=4
        IF (CODE(I).EQ.'3') LL(I)=5
        IF (CODE(I).EQ.'3C') LL(I)=6
        IF (CODE(I).EQ.'5') LL(I)=7
        IF (CODE(I).EQ.'5C') LL(I)=8
        IF (CODE(I).EQ.'8') LL(I)=9
        IF (CODE(I).EQ.'10') LL(I)=10
        IF (CODE(I).EQ.'12') LL(I)=11
        IF (CODE(I).EQ.'15') LL(I)=12
        IF (CODE(I).EQ.'18') LL(I)=13
        IF (LL(I).EQ.0) GO TO 1080
        HAVEDETSFLAG='Y'
1110    CONTINUE
        GO TO 1180
1120    WRITE(*,1130)
1130        FORMAT(10X,'0.....BARE',/,
& 10X,'0C.....BARE+CADMIUM COVER',/,
& 10X,'2.....2 INCH BALL',/,
& 10X,'2C.....2 INCH CADMIUM COVERD BALL',/,
& 10X,'3.....3 INCH BALL',/,
& 10X,'3C.....3 INCH CADMIUM COVERED BALL',/,
& 10X,'5.....5 INCH BALL',/,
& 10X,'5C.....5 INCH CADMIUM COVERD BALL',/,
& 10X,'8.....8 INCH BALL',/,
& 10X,'10.....10 INCH BALL',/,
& 10X,'12.....12 INCH BALL',/,
& 10X,'15.....15 INCH BALL',/,

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& 10X,'18.....18 INCH BALL',/,
& 10X,'6.....2,3,5,8,10,12 INCH BALLS',/,
& 10X,'6B.....BARE,2,3,5,8,10,12 INCH BALLS',/,
& 10X,'7.....2,3,5,8,10,12,18 INCH BALLS',/,
& 10X,'7B.....BARE,2,3,5,8,10,12,18 INCH BALLS',/,/)
GO TO 1080

C
C SELECT APPROPRIATE ROWS FROM MATRIX AND ADJUST
C MATRIX TO UNFOLD PER UNIT LETHARGY
C KK = NUMBER OF DETS, LL(I) = RMTX ROW NUMBER OF THE Ith DET
1180 DO 1190 K=1, KK
      L=LL(K)
      DO 1190 J=1, JJ
1190 ALETH(K,J)=A(L,J)*WDLETH(J)
C
C EITHER CREATE (MAXIET) OR USE AN INITIAL SPECTRUM. FILL SPLI(I).
C
1195 IF(INSFFLAG.EQ.'Y')GO TO 2500
C INSPFLAG.EQ.'Y' MEANS THERE IS ALREADY AN INITIAL SPECTRUM IN SPLI
1200 WRITE(*,1205)JJJ, JJJ
1205 FORMAT(/,'SELECT INITIAL SPECTRUM TYPE:',/,/,
& 10X, '1 .....FLAT',/,
& 10X, '2 .....CF-252 (BARE)',/,
& 10X, '3 .....CF-252 (ROOM RETURN)',/,
& 10X, '4 .....CF-252 (D20-MODERATED)',/,
& 10X, '5 .....NEH D-T SPECTRUM',/,
& 10X, '6 .....AMERICIUM-BERYLLIUM',/,
& 10X, '7 .....DEUTERIUM-BERYLLIUM',/,
& 10X, '8 .....MANUALLY ENTER A SPECTRUM (' ,I2, ' VALUES)',/,
& 10X, '9 .....FILE NAME OF AN INIT SPECTRUM (' ,I2, ' VALUES)',/,
& 9X, '10 .....SEARCH FOR MAXWELLIAN, 1/E SPECTRUM',/)

C
      READ(*,*) INSPANS
      IF (INSPANS.GT.11.OR.INSPANS.LT.1)GO TO 1200
      IF (INSPANS.EQ.1) THEN
        INSPNAM = 'FLAT'
        DATA(SPL1(I), I=1,54)/54*1.0/
        DO 3010 I=1, JJ
3010 SPLI(I)=SPL1(I)
3015 CONTINUE
C
      ELSE IF (INSPANS.EQ.2) THEN
        INSPNAM = '252-CF (BARE)'
        DATA(SPL3(I), I=1,54)/
& 2.06E-12, 6.89E-10, 9.26E-09, 1.38E-07, 6.97E-07,
& 2.15E-06, 6.61E-06, 1.67E-05, 3.53E-05, 1.03E-04,

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&      3.13E-04, 7.13E-04, 1.86E-03, 4.32E-03, 7.64E-03,
&      1.29E-02, 2.16E-02, 3.30E-02, 5.46E-02, 9.64E-02,
&      1.43E-01, 1.90E-01, 2.44E-01, 2.95E-01, 3.31E-01,
&      3.70E-01, 4.12E-01, 4.57E-01, 5.17E-01, 5.92E-01,
&      6.62E-01, 7.23E-01, 8.14E-01, 9.16E-01, 9.81E-01,
&      1.03E+00, 1.06E+00, 1.06E+00, 1.05E+00, 1.00E+00,
&      9.30E-01, 8.04E-01, 6.67E-01, 5.65E-01, 4.06E-01,
&      2.50E-01, 1.60E-01, 9.26E-02, 4.96E-02, 2.76E-02,
&      1.42E-02, 8.25E-03, 4.47E-03, 2.33E-03/

DO 3030 I=1,JJ
3030   SPLI(I)=SPL3(I)
3035   CONTINUE
      ELSE IF (INSPANS.EQ.3) THEN
        INSPNAM = 'CF-ROOM RETURN'
        DATA(SPL4(I), I=1,54)/
&      1.02E+00, 1.02E-01, 1.24E-01, 1.38E-01, 1.55E-01,
&      1.54E-01, 1.53E-01, 1.50E-01, 1.51E-01, 1.65E-01,
&      1.83E-01, 1.88E-01, 1.88E-01, 1.85E-01, 1.79E-01,
&      1.90E-01, 2.42E-01, 3.13E-01, 3.56E-01, 3.71E-01,
&      3.76E-01, 3.99E-01, 4.33E-01, 4.60E-01, 4.75E-01,
&      4.86E-01, 4.92E-01, 4.95E-01, 4.94E-01, 4.87E-01,
&      4.76E-01, 4.62E-01, 4.31E-01, 3.90E-01, 3.56E-01,
&      3.17E-01, 2.74E-01, 2.26E-01, 1.74E-01, 1.18E-01,
&      7.19E-02, 2.56E-02, 5.69E-03, 0.00E+00, 0.00E+00,
&      0.00E+00, 0.00E+00, 0.00E+00, 0.00E+00, 0.00E+00,
&      0.00E+00, 0.00E+00, 0.00E+00, 0.00E+00/
        DO 3040 I=1,JJ
3040   SPLI(I)=SPL4(I)
3045   CONTINUE
      ELSE IF (INSPANS.EQ.5) THEN
        INSPNAM = 'NEH D-T SPECTRUM'
        DATA(SPL5(I), I=1,54)/
&      1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03,
&      1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03,
&      1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03,
&      1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03, 1.59E-03,
&      1.52E-03, 1.38E-03, 1.26E-03, 1.21E-03, 1.27E-03,
&      1.47E-03, 1.87E-03, 2.86E-03, 4.35E-03, 5.56E-03,
&      6.83E-03, 6.97E-03, 5.05E-03, 3.68E-03, 4.27E-03,
&      7.17E-03, 1.90E-02, 4.88E-02, 7.31E-02, 1.06E-01,
&      1.41E-01, 1.53E-01, 1.40E-01, 1.17E-01, 9.42E-02,
&      6.94E-02, 5.11E-02, 3.00E-02, 1.26E-02/
        DO 3050 I=1,JJ
3050   SPLI(I)=SPL5(I)

```

```

3055      CONTINUE
      ELSE IF (INSPANS.EQ.6) THEN
        INSPNAM = 'AMER-BERYL'
        DATA(SPL6(I), I=1,54)/
&      4.57E-27, 1.28E-12, 2.96E-12, 2.88E-12, 2.88E-12,
&      2.88E-12, 2.88E-12, 2.88E-12, 2.88E-12, 2.88E-12,
&      2.88E-12, 2.88E-12, 2.88E-12, 2.88E-12, 9.80E-05,
&      2.02E-04, 2.87E-05, 5.85E-04, 1.94E-03, 1.11E-02,
&      1.98E-02, 2.38E-02, 2.59E-02, 2.77E-02, 2.85E-02,
&      2.94E-02, 2.92E-02, 2.93E-02, 2.98E-02, 3.19E-02,
&      3.62E-02, 4.05E-02, 4.93E-02, 7.00E-02, 8.75E-02,
&      9.98E-02, 1.19E-01, 1.35E-01, 1.57E-01, 1.74E-01,
&      1.91E-01, 1.97E-01, 1.83E-01, 1.62E-01, 1.39E-01,
&      8.88E-02, 5.18E-02, 3.45E-02, 2.06E-02, 1.36E-02,
&      8.18E-03, 4.25E-03, 2.75E-03, 6.60E-04/
      DO 3060 I=1,JJ
3060      SPLI(I)=SPL6(I)
3065      CONTINUE
      ELSE IF (INSPANS.EQ.7) THEN
        INSPNAM = 'DEUT-BERYL'
        DATA(SPL7(I), I=1,54)/
&      2.86E-17, 6.82E-02, 4.66E-01, 8.63E-01, 5.85E-01,
&      4.37E-01, 4.84E-01, 4.44E-01, 4.18E-01, 6.72E-01,
&      8.75E-01, 6.38E-01, 6.37E-01, 6.06E-01, 3.82E-01,
&      3.47E-01, 3.40E-01, 2.45E-01, 2.70E-01, 2.79E-01,
&      1.89E-01, 1.41E-01, 1.02E-01, 7.31E-02, 4.76E-02,
&      3.91E-02, 5.05E-02, 6.28E-02, 8.60E-02, 1.05E-01,
&      8.49E-02, 6.06E-02, 7.40E-02, 1.13E-01, 1.19E-01,
&      1.20E-01, 1.17E-01, 1.44E-01, 1.90E-01, 2.01E-01,
&      1.93E-01, 1.48E-01, 1.14E-01, 1.00E-01, 1.04E-01,
&      9.02E-02, 5.31E-02, 2.82E-02, 1.41E-02, 7.51E-03,
&      3.23E-03, 1.42E-03, 9.93E-04, 5.95E-04/
      DO 3070 I=1,JJ
3070      SPLI(I)=SPL7(I)
3075      CONTINUE
      ELSE IF (INSPANS.EQ.4) THEN
        INSPNAM = 'D2O-MOD CF'
        DATA(SPL11(I), I=1,54)/
&      2.56E-04, 2.95E-03, 4.32E-02, 6.26E-02, 7.73E-02,
&      8.88E-02, 9.92E-02, 1.09E-01, 1.19E-01, 1.27E-01,
&      1.36E-01, 1.42E-01, 1.49E-01, 1.51E-01, 1.47E-01,
&      1.44E-01, 1.41E-01, 1.35E-01, 1.27E-01, 1.17E-01,
&      1.05E-01, 9.78E-02, 8.81E-02, 7.27E-02, 5.37E-02,
&      5.14E-02, 7.27E-02, 9.21E-02, 9.86E-02, 1.02E-01,
&      9.57E-02, 7.56E-02, 7.33E-02, 9.68E-02, 1.19E-01,
&      1.37E-01, 1.48E-01, 1.66E-01, 2.08E-01, 2.16E-01,

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&      1.83E-01, 1.48E-01, 1.29E-01, 1.28E-01, 1.09E-01,
&      7.54E-02, 5.01E-02, 3.04E-02, 1.85E-02, 1.02E-02,
&      4.15E-03, 2.11E-03, 1.58E-03, 9.26E-04/
      DO 3080 I=1,JJ
3080      SPLI(I)=SPL11(I)
3085      CONTINUE
      ELSE IF (INSPANS.EQ.9) THEN
        WRITE(*,1210)
1210      FORMAT(' ENTER THE SPECTRUM FILE OR LIST NAME (21 CHARS MAX)',/,
&      ' MUST HAVE 54 VALUES OF BIN-AVERAGE FLUENCE/LETHARGY',/)
        READ(*,1220) SPFILE
C
C
1220      FORMAT(A21)
        OPEN(10,FILE=SPFILE)
        READ(10,*) (SPLI(I),I=1,JJ)
        CLOSE(10)
        INSPNAM = SPFILE
      END IF
      IF (INSPANS.EQ.8) THEN
        WRITE(*,1225)
1225      FORMAT(' TYPE INITIAL SPECTRUM, 54 VALUES, FLUENCE/LETH')
        READ(*,*) (SPLI(I),I=1,JJ)
        WRITE(*,1210)
1230      READ(*,1220) INSPNAM
      END IF
      IF (INSPANS.EQ.10) THEN
        INSPNAM = 'MAX, 1/E SPT'
C
C SHAPE IS THE SHAPE OF THE HI TEMP PORTION OF THE OF THE MAXWELLIAN
C PEAK. PERTMP IS THE AMOUNT BY WHICH THE MAX TEM IS CHANGED IN
C SEARCHING FOR A BETTER FIT TO THE DATA. POSITIVE VALS SEARCH FOR
C A LOWER TEMP, NEG SEARCHES FOR A HIGHER TEMP, AND ZERO FORCES TEMP TO
C INPUT TEMP, TEMPIJ
C
1300      IF (HAVEMAXFLAG.EQ.'N') GO TO 1400
        WRITE(*,1390) TEMPIJ, SHAPE, PERTMP
1390      FORMAT(' CHANGE MAXWELLIAN TEMP, SHAPE, OR PERTURBATION?',
& 3X,F6.1,F6.1,F6.1)
        READ(*,1050) CHTEMP
        IF (CHTEMP.EQ.'Y') GO TO 1400
        IF (CHTEMP.EQ.'N') GO TO 1450
        IF (CHTEMP.NE.'Y'.AND.CHTEMP.NE.'N') GO TO 1300

1400      WRITE(*,1410)
1410      FORMAT(/,' TYPE MAXWELLIAN TEMP, SHAPE (0-0.5) ',/,

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& ' AND PERTURBATION [W/IN +/- 10% OF TEMP OR 0.0] ',/)
      READ(*,*)TEMPIJ,SHAPE,PERTMP
      HAVEMAXFLAG = 'Y'
      END IF
1415  IF(INSPANS.NE.10)HAVEMAXFLAG = 'N'
      INSPFLAG = 'Y'
      WRITE(*,1420)INSPNAM
1420  FORMAT(/,'INITIAL SPECTRUM SELECTED IS ',A21,/)
C
C
C   INPUT THE INITIAL SPECTRUM FIT PARAMETERS
C   REPEAT THE FIT VALUES AND ASK TO CHANGE =>2ND TIME THROUGH
C
1450  IF(HAVEFITFLAG.EQ.'N')GO TO 1500
1475  WRITE(*,1480)
1480  FORMAT(/,' CHANGE FIT PARAMETERS ?')
      WRITE(*,1485)TSTPE,SMO,CAL,ITRTST,ITRMAX
1485  FORMAT(/,
& 3X,'                                END TEST(%) = ',F3.1,/,
& 3X,'                                SMOOTHING FACTOR = ',F3.1,/,
& 3X,'                                CALIBRATION FACTOR = ',F3.1,/,
& 3X,'ITERATIONS BEFORE ERROR TEST = ', I4,/,
& 3X,'MAXIMUM NUMBER OF ITERATIONS = ',I4,/)
      READ(*,1050)CHFIT
      IF(CHFIT.NE.'Y'.AND.CHFIT.NE.'N')GO TO 1475
      IF(CHFIT.EQ.'Y')GO TO 1500
      IF(CHFIT.EQ.'N')GO TO 1525
C
C   REQUEST FIT PARAMETER3:
C
C   TSTPER AND TESTPE IS THE ACCEPTABLE ERROR ON THE FIT. SMO IS THE
C   SMOOTHING FACTOR. ITRTST IS THE NUMBER OF ITER BEFORE TESTING TSTPER,
C   ITRMAX IS THE MAX NUMBER OF ITERATIONS BEFORE TERMINATING.
C
1500  WRITE(*,1510)
1510  FORMAT(
& /,' TYPE: END TEST(%)',',
& /,'          SMOOTHING FACTOR (TYP 0-0.5), ',
& /,'          CALIBRATION FACTOR, ',
& /,'          ITERATIONS BEFORE ERROR TEST (INTEGER: TYP 1-100), ',
& /,'          AND MAXIMUM NUMBER OF ITERATIONS (TYP 100-1000)',/,

      READ(*,*)TSTPE,SMO,CAL,ITRTST,ITRMAX
C   IF(HAVEMAXFLAG.EQ.'N'.AND.ITRMAX.EQ.0)GO TO 1500
      WRITE(*,1485)TSTPE,SMO,CAL,ITRTST,ITRMAX
      HAVEFITFLAG='Y'

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```

C          PERTHM=1.0+20*PERSLP ! THE AMOUNT BY WHICH THERM BIN OF
C                                     MAXIET SPT IS CHANGED IN SEARCHING FOR
C
C          A BETTER FIT TO THE DATA.
C          PERE=1.0+10*PERSLP ! THE AMNT BY WHICH MAG OF THE 1/E PART
C                                     MAXIET SPT IS CHANGED IN SEARCHING FOR
C
C          A BETTER FIT TO THE DATA.
C          TSTPER=(KK*TSTPE**2)/10000
1520      IF (HAVEMAXFLAG.EQ.'N'.AND.ITRMAX.EQ.0)GO TO 1820
C          TEMPI=TEMPIJ
C          SLOPEI=SLOPEJ
C          THERMI=THERMJ
C          GO TO 1525
C
C          IF ALREADY HAVE AN INITIAL SPECTRUM, ASK TO CHANGE IT
2500      WRITE(*,2505)
2505      FORMAT(/,' CHANGE THE INITIAL SPECTRUM?',/)
C          READ(*,1050)CHISPC
C          IF(CHISPC.NE.'Y'.AND.CHISPC.NE.'N')GO TO 2500
C          IF(CHISPC.EQ.'Y')GO TO 1200
C          IF(CHISPC.EQ.'N')GO TO 1450
C
C          AN INITIAL SPECTRUM IS CHOSEN, (OR IS YET TO BE CALCULATED IF
C          A MAX,1/E IS TO BE FOUND BECAUSE MAX-1/E INIT SPT
C          DEPENDS ON THE BALL DATA), AND FIT PARAMETERC ARE SET.
C          NOW INPUT THE BALL DATA:
C
1525      IF (HAVEBALLDATA.EQ.'N'.OR.CHDET.EQ.'Y')GO TO 1540
1530      WRITE(*,1535)
1535      FORMAT(/,' CHANGE BALL DATA?',/)
C          READ(*,1050)CHBCE
C          IF(CHBCE.EQ.'Y')GOTO 1540
C          IF(CHBCE.EQ.'N')GO TO 1590
C          GO TO 1530
1540      WRITE(*,1545)KK
1545      FORMAT(/,' DO YOU HAVE THE BALL DATA IN A FILE? (Y OR N)'/
& ' (FORMATTED AS COUNT, % ERROR; NEED ' ,12,' PAIRS OF VALUES)',/)
C          READ(*,1050) BALLANS
C          IF(BALLANS.NE.'Y'.AND.BALLANS.NE.'N')GO TO 1540
C          IF(BALLANS.EQ.'Y') THEN
C              WRITE(*,1550)
1550      FORMAT(/,' ENTER FILE NAME OF BALL DATA. ',/)
C          READ(*,1220) BALFILE
C          OPEN(4,FILE=BALFILE)
C          DO 1555 I=1,KK

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      READ(4,*) BCE(I),ERRBCE(I)
1555      CONTINUE
      CLOSE(4)
      ELSE
      BALFILE='MAN INPUT'
1560      DO 1570 I=1, KK
          L=LL(I)
          WRITE(*,1565) BALL(L)
1565          FORMAT(' TYPE ',A8,' BONNER SPHERE COUNT, % ERROR')
1570          READ(*,*) BCE(I),ERRBCE(I)
      ENDIF
      HAVEBALLDATA='Y'
C      SUM ERRORS, MAKE DEAD TIME CORRECTION TO BALL COUNTS
      SUMWHT=0
      DO 1575 I=1, KK
          SUMWHT=SUMWHT+ERRBCE(I)
1575      BCE(I)=BCE(I)/(1.0-BCE(I)*DEAD)
C      CALCULATE BALL COUNT ERROR WEIGHTS
      DO 1580 I=1, KK
1580      WHTBCE(I)=SUMWHT/(KK*ERRBCE(I))
C
C      THERE IS NOW BALL DATA. NEXT CALCULATE MAXWELLIAN, 1/E
C      SPECTRUM IF REQUIRED OR GO ON TO SPUNIT
C
1590      IF(HAVEMAXFLAG.EQ.'N')GO TO 1820
C
C*****BEGINNING OF MAXIET ALGORITHM*****
C
C      MAXIET IS AFTER BALL DATA READ-IN BECAUSE BALL DATA IS USED TO
C      CALCULATE THE MAX, 1/E SPECTRUM. MAXIET HAS BEEN CHANGED BECAUSE
C      PROGRAMMING SYNTAXES WERE WRONG (i+1 VALUES WERE CALLED WHEN THE
C      ARRAY HAD ONLY i ELEMENTS.
C
C      INITIALIZE FIT PARAMETERS
      ERRORE=123456789
      TEMP=TEMPI
      ERRORM=ERRORE
1600      SLOPE=SLOPEI
      THERM=THERMI
C      CALCULATE MAXWELLIAN SPECTRUM
      SPMX=0
      DO 1610 I=1, JJJ
          SPLMAX(I)=(EEND(I)**1.5)*EXP(-EEND(I)/TEMP)
          IF (SPLMAX(I).GT.SPMX) SPMX=SPLMAX(I)
          IF (SPMX.EQ.SPLMAX(I))GO TC 1610
          SPLMAX(I)=SPMX**SHAPE*SPLMAX(I)**(1.0-SHAPE)

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        IF(SPLMAX(I).LT.SPLMAX(I-1)*SHP) SPLMAX(I)=SPLMAX(I-1)*SHP
1610    CONTINUE
C      CALCULATE 1/E INITIAL SPECTRUM
        ERROR=123456789
        ERRORE=123456789
        HGTE=SPMX*PERE
1620    ERRORT=ERROR
1630    HGTE=HGTE/PERE
1640    DO 1650 I=1,JJJ
1650        SPLM(I)=HGTE*EEND(I)**SLOPE
C      COMBINE MAXWELLIAN AND 1/E SPECTRA
        DO 1660 I=1,JJJ
            IF(SPLM(I).LT.SPLMAX(I))GO TO 1670
            SPLM(I)=(SPLM(I)+SPLMAX(I))*0.5
1660    CONTINUE
        GO TO 1630
1670    DO 1680 J=1,JJJ
1680        SPLM(J)=SPLMAX(J)
C      ADJUST THERMAL ENERGY BIN
        SPLM(1)=SPLM(2)*THERM
C      CALCULATE SPHERE RESPONSES AND SUM FROM SPECTRUM
        DO 1690 M=1,KK
            BCC(M)=0
            DO 1690 J=1,JJ
1690        BCC(M)=BCC(M)+ALETH(M,J)*(SPLM(J)*SPLM(J+1))**.5
C
C      CALCULATE SUMS OF SPHERE DATA
        SUMBCE=0
        SUMBCC=0
        DO 1700 I=1,KK
            SUMBCE=SUMBCE+BCE(I)
            SUMBCC=SUMBCC+BCC(I)
1700
C      NORMALIZE CALCULATED SPHERE RESPONSES
C      TO EXPERIMENTAL DATA
        RNORM=SUMBCE/SUMBCC
        DO 1710 I=1,KK
1710        BCC(I)=BCC(I)*RNORM
C      CALCULATE ERROR ON FIT
        ERROR=0
        DO 1720 I=1,KK
            ERR=(BCC(I)-BCE(I))/BCE(I)
1720        ERROR=ERROR+WHTBCE(I)*ERR*ERR
            IF(ERROR.LT.ERRORT)GO TO 1620
            HGTE=HGTE*PERE
1730        IF(ERROR.GE.ERRORE)GO TO 1740
C      SAVE BEST VALUES OF FIT PARAMETERS

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```

        ERRORE=ERRORT
        HGTEE=HGTE
        THERME=THERM
        SLOPEE=SLOPE
        MX=0
C      CHANGE SLOPE
1740    IF(SLOPE.LT.SLPMAX) SLOPE=SLOPE+PERSLP
        IF(MX.EQ.1) SLOPE=SLOPEE
C      CHANGE THERMAL BIN
        THERM=THERME*PERTHM
        IF(THERM.GE.THMMAX) GO TO 1750
        IF(MX.EQ.0) THERM=THERME
        MX=MX+1
        IF(MX.GT.10) GO TO 1750
        HGTE=HGTE*PERE*(1.0+10*PERSLP)
        IF(MX.EQ.1) HGTE=HGTEE*PERE*PERTHM
C      RESET ERROR, SEARCH FOR BETTER FIT PARAMETERS
        ERRORT=123456789
        GO TO 1640
C      CALCULATE ERROR ON FIT
1750    PERROR=100*(ERRORE/KK)**.5
C      WRITE BEST VALUES OF FIT PARAMETERS TO TERMINAL
        WRITE(*,*)TEMP,SHAPE,HGTEE,SLOPEE,THERME,PERROR
        IF(ERRORE.GE.ERRORM) GO TO 1760
C      SAVE BEST VALUES OF FIT PARAMETERS
        TEMPM=TEMP
        HGTEM=HGTEE
        THERMM=THERME
        SLOPEM=SLOPEE
        ERRORM=ERRORE
C      CHANGE MAXWELLIAN TEMPERATURE IF REQUIRED
        IF(PERTMP.EQ.0) GO TO 1760
        TEMP=TEMP-PERTMP
C      RETURN AND SEARCH FOR BETTER PARAMETERS IF
C      MAXWELLIAN TEMP IS IN RANGE
        IF(TEMP.GT.PERTMP) GO TO 1600
1760    CONTINUE
C      IF FINAL PARAMETERS EQUAL INITIAL PARAMETERS CHANGE
C      INITIAL PARAMETERS AND CONTINUE SEARCH
        A1=0
        IF(SLOPEM.EQ.SLOPEI.AND.SLOPEI.GT.SLPMIN) A1=1
        A2=0
        IF(THERMM.EQ.THERMI.AND.THERMI.GE.THMMIN) A2=1
        A3=0
        IF(TEMPM.EQ.TEMPI.AND.TEMPM.LT.TEMPIJ+10*PERTMP) A3=1
        IF(PERTMP.EQ.0) A3=0

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      IF (A1.EQ.1) SLOPEI=SLOPEI-10*PERSLP
      IF (A2.EQ.1) THERMI=THERMI/PERTHM**3
      IF (A3.EQ.1) TEMPI=TEMPI+3*PERTMP
      IF (A1+A2+A3.GT..5) GO TO 1590
C    IF NOT, CALCULATE INITIAL SPECTRUM WITH BEST PARAMETERS
      SPMX=0
      DO 1770 I=1, JJJ
        SPLMAX(I)=(EEND(I)**1.5)*(EXP(-EEND(I)/TEMPM))
        IF (SPLMAX(I).GT.SPMX) SPMX=SPLMAX(I)
        IF (SPMX.EQ.SPLMAX(I)) GO TO 1770
        SPLMAX(I)=SPMX**SHAPE*SPLMAX(I)**(1.0-SHAPE)
        IF (SPLMAX(I).LT.SPLMAX(I-1)*SHP) SPLMAX(I)=SPLMAX(I-1)*SHP
1770    CONTINUE
      DO 1780 I=1, JJJ
        SPLM(I)=HGTEM*EEND(I)**SLOPEM
1780      DO 1790 I=1, JJJ
        IF (SPLM(I).LT.SPLMAX(I)) GO TO 1800
        SPLM(I)=(SPLM(I)+SPLMAX(I))*0.5
1790      CONTINUE
1800      DO 1810 J=1, JJJ
1810        SPLM(J)=SPLMAX(J)
        SPLM(1)=SPLM(2)*THERMM
C    CALCULATE BIN-AVERAGE FLUENCES BY AVERAGING ENDPOINT VALUES:
      DO 1815 J=1, JJ
1815        SPLI(J)=(SPLM(J)*SPLM(J+1))**.5
C
C***** COMPLETION OF MAXIET ALGORITHM *****
C
C    TRANSFORM MATRIX TO CONSTANT INITIAL SPECTRUM
1820      DO 1825 K=1, KK
        DO 1825 J=1, JJ
1825        ALETH(K, J)=ALETH(K, J)*SPLI(J)
        DO 1830 I=1, JJ
1830        SPL(I)=1
C    CALCULATE SPHERE RESPONSES AND SUM FROM INITIAL SPECTRUM
      DO 1840 M=1, KK
        TC(M)=0
        DO 1840 J=1, JJ
1840        BCC(M)=BCC(M)+ALETH(M, J)*SPL(J)
        IF (HAVEMAXFLAG.EQ.'N'.AND.ITRMAX.EQ.0) GO TO 2000
C    NORMALIZE CALCULATED SPHERE RESPONSES AND INITIAL
C    SPECTRUM TO EXPERIMENTAL DATA
      SUMBCC=0
      SUMBCE=0
      DO 1850 I=1, KK
        SUMBCE=SUMBCE+BCE(I)

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1850      SUMBCC=SUMBCC+BCC(I)
      RNORM=SUMBCE/SUMBCC
      DO 1860 I=1, KK
1860      BCC(I)=BCC(I)*RNORM
      SUMBCC=SUMBCE
      DO 1870 I=1, JJ
1870      SPL(I)=SPL(I)*RNORM
C      CALCULATE ERROR ON FIT
      ERROR=0
      DO 1880 I=1, KK
          ERR=(BCC(I)-BCE(I))/BCE(I)
1880      ERROR=ERROR+WHTBCE(I)*ERR*ERR
          IF(ITRMAX.EQ.0)GO TO 2000
          ITER=0
1890      ERRORU=ERROR
C
C***** BEGINNING OF SPUNIT UNFOLDING ALGORITHM *****
C      BASED ON DOROSHENKO'S EQ 7 (NIM, 33, 296-304, 1977)
C
C      BCC - CALCULATED BALL COUNT ON Ith ITERATION [C(i,j)]
C      BCE - MEASURED BALL COUNT [C(0,j)]
C      ALETH - RESPONSE FUNCTION
C      SPL - CALCULATED FLUENCE ON Ith ITERATION [F(i,j)]
C      SPLL - CALCULATED FLUENCE ON I+1 ITERATION [F(i+1,j)]
C      SS - SUMMATION W/IN ITERATED SUMMATION
C
      IF(ITER.GT.0)GO TO 1905
      DO 1900 J=1, JJ
          SS(J)=0
          DO 1900 I=1, KK
1900      SS(J)=SS(J)+ALETH(I,J)/BCE(I)
1905      DO 1940 K=1, ITRTST
          ITER=ITER+1
          DO 1910 J=1, JJ
              SPLL(J)=0
              DO 1910 I=1, KK
                  SPLL(J)=SPLL(J) + SPL(J)*ALETH(I,J)/(SS(J)*BCC(I))
C      TO AVOID UNDERFLOW PROBLEMS :
1910      IF(SPLL(J).LT.1.0E-37) SPLL(J)=0.0
C      NOW THE i+1 ITERATION (SPLL) BECOMES THE ith (SPL), SMOOTHING
C      ALONG THE WAY IF NECESSARY (SMO), BUT
C      DO NOT SMOOTH THE THERMAL BINS:
          SPL(1)=SPLL(1)
          SPL(2)=SPLL(2)
          DO 1920 J=3, JJ
1920      SPL(J)=(SPLL(J-1)*SMO+SPLL(J)+SPLL(J+1)*SMO)/(1+2*SMO)

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1925      DO 1930 M=1, KK
          BCC(M)=0
          DO 1930 J=1, JJ
1930      BCC(M)=BCC(M)+ALETH(M, J)*SPL(J)
1940      CONTINUE
C
C ***** END OF SPUNIT UNFOLDING ALGORITHM *****
C
C      CALCULATE ERROR ON FIT
          ERROR=0
          DO 1995 I=1, KK
              ERR=(BCC(I)-BCE(I))/BCE(I)
1995      ERROR=ERROR+WHTBCE(I)*ERR*ERR
C      TEST FOR COMPLETION, CONTINUE IF NOT COMPLETE
          STUF=SQRT(ERROR/KK)*100
          WRITE(*,1961) ITER, STUF
1961      FORMAT(1X, 'ITERATIONS= ', I8, '      ERROR= ', F10.6, /)
          IF (ERROR/ERRORU.LT.TSTRAT.AND.ITER+ITRTST.LE.ITRMAX
              & .AND.ERROR.GT.TSTPER) GO TO 1890
C      IF COMPLETE, DO INVERSE TRANSFORM OF SPECTRUM AND MATRIX
2000      DO 2005 I=1, JJ
2005      SPL(I)=SPL(I)*SPLI(I)
C      CALCULATE CUTPUT VALUES
          IF (HAVEMAXFLAG.EQ.'N'.AND.ITRMAX.EQ.0) GO TO 2030
2015      HGTEM=0.5*HGTEM/SPMX
          SUMERR=0
          DO 2020 I=1, KK
              PCTERR(I)=100*(BCC(I)-BCE(I))/BCE(I)
2020      SUMERR=SUMERR+PCTERR(I)*PCTERR(I)
          PERROR=(SUMERR/KK)**.5
2030      SUMSPC=0
          SUMRAD=0
          SUMREM=0
          SUMEXS=0
          SUMHEFF=0
          SUMHEFFR=0
          SUMAMD=0
          SUMAMH2=0
          SUMAMH6=0
          SUMEFF=0
          SUMEFFI=0
          DO 2035 I=1, JJ
              SPL(I)=SPL(I)*CAL
              SPC(I)=SPL(I)*WDLETH(I)
              SUMSPC=SUMSPC+SPC(I)
              SUMREM=SUMREM+CREM(I)*SPC(I)

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RAD(I)=CRAD(I)*SPC(I)
SUMRAD=SUMRAD+RAD(I)
SUMEXS=SUMEXS+CE(I)*SPC(I)
SUMEFF=SUMEFF+CEFF(I)*SPC(I)
SUMEFFI=SUMEFFI+CEFFI(I)*SPC(I)
PHEFF(I)=CHEFF(I)*SPC(I)
SUMHEFF=SUMHEFF+PHEFF(I)
SUMHEFFR=SUMHEFFR+CHEFFR(I)*SPC(I)
SUMAMD=SUMAMD+CAMD(I)*SPC(I)
SUMAMH2=SUMAMH2+CAMH2(I)*SPC(I)
PAMH6(I)=CAMH6(I)*SPC(I)
SUMAMH6=SUMAMH6+PAMH6(I)
IF(REM(I).LT.1.0E-37) REM(I)=0.0
IF(RAD(I).LT.1.0E-37) RAD(I)=0.0
2035 CONTINUE
QF=SUMREM/SUMRAD ! (Avg QF from D and H in pSv/pGy)
QF2=SUMAMH2/SUMAMD ! (Avg effective QF* - ICRP 26 based)
QF6=SUMAMH6/SUMAMD ! (Avg effective QF* - ICRP 60 based)
AVEEN=SUMEXS-(CE(1)*SPC(1)) !DON'T COUNT THE
AVEEN=AVEEN/(SUMSPC-SPC(1)) !TWO MOST THERM BINS
AVEENW=SUMEXS/SUMSPC !WITH THE THERMAL GROUP
HF = SUMREM/SUMSPC
HF2 = SUMAMH2/SUMSPC
HF6 = SUMAMH6/SUMSPC
DO 2040 I=1,JJ
PHEFF(I)=100*(PHEFF(I)/SUMHEFF)
PAMH6(I)=100*(PAMH6(I)/SUMAMH6)
2040 CONTINUE
C SET UNCOMPUTED PARAMETERS TO 0, WRITE PARAMETERS TO TERMINAL
IF(HAVEMAXFLAG.EQ.'N')THERMM=0
IF(HAVEMAXFLAG.EQ.'N')TEMPM=0
IF(HAVEMAXFLAG.EQ.'N')HGTEM=0
IF(HAVEMAXFLAG.EQ.'N')SLOPEM=0
IF(HAVEMAXFLAG.EQ.'N'.AND.ITRMAX.EQ.0)PERROR=0
IF(ITRMAX.EQ.0)ITER=0
WRITE(*,2043)TEMPM,SHAPE,HGTEM,SLOPEM,THERMM,PERROR,ITER
2043 FORMAT(
& ' TEMP = ',F4.1,' SHAPE = ',F4.1,' HGTE = ',F4.1,
& ' SLOPE = ',F4.1,/,
& ' THERM = ',F4.1,' PCT ERR = ',F4.1,' ITER = ',F4.1,/)
C DECIDE IF RESULTS ARE WORTH KEEPING
2045 WRITE(*,2050)
2050 FORMAT(' SAVE THESE RESULTS?',/)
READ(*,1050)SAVE
IF(SAVE.NE.'Y'.AND.SAVE.NE.'N')GO TO 2045
IF(SAVE.EQ.'N')GO TO 2190

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        IF(SAVE.EQ.'Y')GO TO 2055
        GO TO 2045
C      WRITE THE OUTPUT TO THE FILE TO BE DESIGNATED
2055    WRITE(*,2057)
2057    FORMAT(/' ENTER THE OUTPUT FILE NAME',/)
        READ(*,1220)FILNAM
        FILNAMH = FILNAM
        WRITE(*,2058)
2058    FORMAT(/' ENTER A HEADER OF FILE TO BE SAVED [<50 CHARS] ',/)
        READ(*,2059)HEAD
2059    FORMAT(A50)
        HEAD=FILNAMH//': '//HEAD
        OPEN(1,FILE=FILNAM,STATUS='NEW')
        IF(ITRMAX.EQ.0)GO TO 2080
        WRITE(1,2065),HEAD, BALFILE,ITRTST, TSTPE
2065    FORMAT(' ***** AFITBUNKI ',
& ' *****',/,/,
& ' OUTPUT FILE SAVED AS ',A50,/,
& ' BALL DATA FILE IS ', A50,/,
& ' ITERATIONS BETWEEN END TESTS WAS ', I3,/,
& ' SPECIFIED END TEST WAS ',F3.1,'% ',/)

        WRITE(1,2070) INSPNAM, TEMPM, SHAPE, CAL, SMO, PERROR, ITER
2070    FORMAT(4X, 'INITIAL          MAXWELL          CALIB.'
& '    SMOOTH    PER CENT NO. OF',/,
& 4X, 'SPECTRUM      TEMP, SHAPE    FACTOR    FACTOR    ERROR'
& '    ITERATIONS',
& /, 4X, A14, F6.2, ' ', F4.2, F11.4, 2(F10.4), I9)
        GO TO 2100

2080    WRITE(1,2085), HEAD, BALFILE
2085    FORMAT(' ***** AFITBUNKI ',
& ' *****',/,/,
& ' OUTPUT FILE SAVED AS ',A50,/,
& ' BALL DATA FILE IS ', A50,/,
& ' CALCULATIONS BASED ON INITIAL SPECTRUM, DATA NOT UNFOLDED',/,/)

        WRITE(1,2090) INSPNAM, TEMPM, SHAPE, HGTEM, SLOPEM, THERMM, PERROR, CAL
2090    FORMAT(
& 2X, 'INITIAL          MAXWELL    1/E X    LETH    THERMAL PERCENT'
& '    CALIB.',/,
& 2X, 'SPECTRUM      TEMP, SHAPE    FACTOR    SLOPE    FACTOR    ERROR '
& '    FACTOR',
& /, 2X, A14, F4.2, ' ', F4.2, F9.4, F8.4, F8.4, 2X, F7.3, F11.2)
2100    WRITE(1,2110)

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2110  FORMAT(1X/10X,9HDETECTORS,10X,8HMEASURED,5X,
& 12H  CALCULATEL,5X,7HPERCENT,/
& 30X,6HCOUNTS,10X,6HCOUN S,6X,10HDIFFERENCE)
      DO 2120 I=1,KK
        L=LL(I)
2120  WRITE(1,2130) BALL(L),BCE(I),BCC(I),PCTERR(1)
2130  FORMAT(10X,A10,1X,0PF15.3,0PF16.3,0PF15.3)
      WRITE(1,2140) SUMSPC,AVEENW,AVEEN,SUMRAD,SUMREM,QF,HF,SUMHEFF,
& SUMHEFFR,SUMEFF,SUMEFFI,SUMAMD,SUMAMH2,QF2,HF2,SUMAMH6,QF6,HF6
2140  FORMAT(/,
& 15X,'          NEUTRON SPECTRUM DATA ',/,/,
& 7X,'TOTAL FLUENCE (F) = ',1PE11.3,3X,
& 'neut/cm^2',/,
& 7X,'AVG ENERGY (INCL THERM, E=> 1E-11 MeV) = ',1PE11.3,3X,
& 'MeV',/,
& 7X,'AVG ENERGY (LESS THERM, E=> 0.414 eV) = ',1PE11.3,3X,
& 'MeV',/,/,
& 15X,' TRADITIONAL DOSE/DOSE EQUIVALENT QUANTITIES',/,/,
& 7X,'DOSE (D) {ICRP 21} = ',1PE11.3,3X,
& 'pGy',/,
& 7X,'DOSE EQUIVALENT (H) {ICRP 21} = ',1PE11.3,3X,
& 'pSv',/,
& 7X,'AVERAGE QUALITY FACTOR (Q) {ICRP 21} = ',0PF7.3,3X,
& ' pSv/pGy',/,
& 7X,'DOSE EQUIVALENT/FLUENCE (H/F) = ',1PE11.3,3X,
& 'pSv-cm^2/n',/,/,
& 14X,'TRADITIONAL EFFECTIVE DOSE EQUIVALENT QUANTITIES ',/,/,
& 7X,'EFFECTIVE DOSE EQUIVALENT (HE) {ICRP 26/51}',/,/,
& 7X,'(PRE-1985 QUALITY FACTORS) AP EXPOSURE = ',1PE11.3,3X,
& 'pSv',/,
& 7X,'          ROT EXPOSURE = ',1PE11.3,3X,
& 'pSv',/,/,
& 15X,'          VOLUMETRIC DOSE QUANTITIES ',/,/,
& 7X,'EFFECTIVE DOSE (E) {ICRP 60} AP EXPSR = ',1PE11.3,3X,
& 'pSv',/,
& 7X,'          ISOTROPIC EXPOSURE = ',1PE11.3,3X,
& 'pSv',/,/,
& 15X,' PRE-1985 ENVIRONMENTAL MONITORING QUANTITIES ',/,/,
& 7X,'AMBIENT DOSE (D*[10]) {ICRP 26} = ',1PE11.3,3X,
& 'pGy',/,
& 7X,'AMBIENT DOSE EQUIV (H*[10]) {ICRP 26} = ',1PE11.3,3X,
& 'pSv',/,
& 7X,'EFF AMB QUALITY FACTOR (Q*) {ICRP 26} = ',0PF7.3,3X,
& ' pSv/pGy',/,
& 7X,'AMB DOSE EQIV/FLUENCE (H*/F) {ICRP 26} = ',1PE11.3,3X,
& 'pSv-cm^2/n',/,/,

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& 15X,'POST-1985 ENVIRONMENTAL MONITORING QUANTITIES ',/,/,
& 7X,'AMB DOSE EQUIVALENT (H*[10]) (ICRP 60) = ',1PE11.3,3X,
& 'pSv',/,
& 7X,'EFF AMB QUALITY FACTOR (Q*) (ICRP 60) = ',0PF7.3,3X,
& 'pSv/pGy',/,
& 7X,'AMB DOSE EQIV/FLUENCE (H*/F) (ICRP 60) = ',1PE11.3,3X,
& 'pSv-cm^2/n',/,/)

C
      WRITE(1,2145),HEAD
2145   FORMAT(' ***** AFITBUNKI ',
& ' *****',/,/,
& ' OUTPUT FILE SAVED AS ',A50,/,)

      WRITE(1,2150)
2150   FORMAT('  BIN      ENERGY      FLUENCE      FLUENCE      '
& 'DOSE      HsubE [AP]  H*(10) [60]',/,
& '  NO.      MAX (MeV)  n/cm2      n/cm2/leth pGy      '
& '% of Tot    % of Tot')
      DO 2160 I=1,JJ
2160   WRITE(1,2170) I,EEND(I+1),SPC(I),SPL(I),RAD(I),
& PHEFF(I),PAMH6(I)
2170   FORMAT(2X,I4,2X,4(1PE11.3),1PE10.2,1X,1PE11.2)
      CLOSE(1)
2190   CONTINUE

C  FOLLOWING WRITES LIST OF FLUENCES/LETHARGY (SPL(I)) FOR EXTERNAL
C  PLOTTING ( THIS DOES NOT WRITE THE CORRESPONDING ENERGIES )
3999   WRITE(*,4000)
4000   FORMAT(/,' DO YOU WANT TO WRITE A LIST FILE OF FLUENCE/LETH?',/)
      READ(*,1050) TKANS
      IF (TKANS.NE.'Y' .AND. TKANS.NE.'N') GOTO 3999
      IF (TKANS.EQ.'Y') THEN
        WRITE(*,4020)
4020   FORMAT(/,' ENTER THE LIST NAME FOR OUTPUT. (< 8 CHARACTERS)',/)
        READ(*,4021) LISTNM
4021   FORMAT(A8)
        WRITE(*, 4041)LISTNM
        TKFILENM=LISTNM//'.TK.OUT'
        WRITE(*, 4042)TKFILENM
        OPEN(10,FILE=TKFILENM)
        WRITE(10,4030) LISTNM
4030   FORMAT(A12,':')
C  SPL(I) IS THE FLUENCE/UNIT LETHARGY IN THE Ith BALL
      DO 4035 I=1, JJ-1
4035   WRITE(10,4040) SPL(I)
      CONTINUE

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        WRITE(10,4045) SPL(JJ)
4040  FORMAT(F8.4,"","")
4045  FORMAT(F8.4,/,/)
        CLOSE(10)
4041  FORMAT(/,' TK LIST NAME IS: ',A8)
4042  FORMAT(/,' TK FILE NAME IS: ',A60)
        ENDIF
C  END FLUENCE/LETHARGY FILE PRINTOUT
C
C  SET FLAGS, RETURN FOR ANOTHER SPECTRUM IF DESIRED
        CHDF='N'
        CHNUM='N'
2200  WRITE(*,2210)
2210  FORMAT(/,' LAST SPECTRUM?',/)
        READ(*,1050)LASTSP
        IF(LASTSP.EQ.'N')GO TO 2220
        IF(LASTSP.EQ.'Y')GO TO 2240
        GO TO 2200
2220  WRITE(*,2230)
2230  FORMAT(/,' CHANGE DETECTORS?',/)
        READ(*,1050)CHMTX
        IF(CHMTX.EQ.'N')GO TO 1180
        IF(CHMTX.EQ.'Y') THEN
            HAVEDETSFLAG = 'N'
            HAVEBALLDATA = 'N'
            GO TO 1030
        END IF
        GO TO 2220
2240  CLOSE(1)
9999  END

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Appendix B: AFITBUNKI's UTA54 Response Matrix

AFITBUNKI'S UTA54 Response Matrix collapsed from Hertel
and Davidson's 171 Energy-Group Response Matrix [9:509]

Grp	Max Energy	Bare	Bare w/Cd	2"	2" w/Cd	3"	3" w/Cd
54	1.492E+01	5.525E-05	6.480E-05	4.735E-04	8.360E-04	4.170E-03	5.650E-03
53	1.419E+01	4.605E-05	5.540E-05	4.735E-04	8.240E-04	4.322E-03	5.768E-03
52	1.284E+01	5.380E-05	6.230E-05	5.060E-04	8.290E-04	4.590E-03	5.940E-03
51	1.221E+01	6.130E-05	6.835E-05	5.435E-04	8.160E-04	4.905E-03	6.075E-03
50	1.105E+01	6.290E-05	6.765E-05	5.910E-04	7.845E-04	5.345E-03	6.240E-03
49	1.000E+01	5.870E-05	6.190E-05	6.485E-04	7.880E-04	5.895E-03	6.600E-03
48	9.048E+00	7.567E-05	7.837E-05	7.880E-04	9.173E-04	7.107E-03	7.787E-03
47	7.788E+00	8.385E-05	8.645E-05	9.445E-04	1.075E-03	8.475E-03	9.170E-03
46	7.047E+00	7.604E-05	7.896E-05	1.119E-03	1.266E-03	1.010E-02	1.081E-02
45	6.065E+00	9.387E-05	9.692E-05	1.422E-03	1.585E-03	1.245E-02	1.325E-02
44	4.966E+00	8.950E-05	9.285E-05	1.640E-03	1.815E-03	1.420E-02	1.505E-02
43	4.493E+00	9.770E-05	1.010E-04	1.860E-03	2.040E-03	1.590E-02	1.680E-02
42	4.066E+00	1.205E-04	1.235E-04	2.255E-03	2.455E-03	1.865E-02	1.955E-02
41	3.329E+00	1.420E-04	1.455E-04	2.690E-03	2.910E-03	2.180E-02	2.270E-02
40	3.012E+00	1.647E-04	1.687E-04	3.107E-03	3.350E-03	2.463E-02	2.563E-02
39	2.592E+00	1.976E-04	2.021E-04	3.620E-03	3.880E-03	2.808E-02	2.910E-02
38	2.307E+00	2.249E-04	2.306E-04	4.145E-03	4.440E-03	3.141E-02	3.249E-02
37	2.019E+00	2.485E-04	2.560E-04	4.695E-03	5.020E-03	3.485E-02	3.600E-02
36	1.827E+00	2.603E-04	2.660E-04	5.333E-03	5.723E-03	3.870E-02	3.990E-02
35	1.572E+00	2.555E-04	2.600E-04	6.025E-03	6.410E-03	4.270E-02	4.380E-02
34	1.423E+00	2.583E-04	2.587E-04	6.790E-03	7.067E-03	4.683E-02	4.773E-02
33	1.225E+00	2.518E-04	2.523E-04	7.962E-03	8.270E-03	5.287E-02	5.382E-02
32	1.003E+00	2.529E-04	2.555E-04	9.077E-03	9.419E-03	5.825E-02	5.921E-02
31	9.072E-01	2.615E-04	2.675E-04	9.890E-03	1.025E-02	6.190E-02	6.285E-02
30	8.209E-01	2.767E-04	2.783E-04	1.097E-02	1.137E-02	6.653E-02	6.747E-02
29	7.065E-01	3.040E-04	3.057E-04	1.240E-02	1.280E-02	7.207E-02	7.300E-02
28	6.081E-01	3.400E-04	3.420E-04	1.365E-02	1.405E-02	7.670E-02	7.760E-02
27	5.507E-01	3.820E-04	3.840E-04	1.465E-02	1.510E-02	8.030E-02	8.120E-02
26	4.979E-01	4.420E-04	4.430E-04	1.570E-02	1.620E-02	8.380E-02	8.470E-02
25	4.505E-01	5.340E-04	5.340E-04	1.690E-02	1.730E-02	8.730E-02	8.820E-02
24	4.076E-01	6.785E-04	6.790E-04	1.800E-02	1.850E-02	9.075E-02	9.165E-02
23	3.688E-01	1.174E-03	1.176E-03	1.993E-02	2.048E-02	9.588E-02	9.663E-02
22	2.985E-01	2.578E-03	2.581E-03	2.256E-02	2.311E-02	1.018E-01	1.028E-01
21	2.472E-01	2.875E-03	2.857E-03	2.470E-02	2.527E-02	1.072E-01	1.080E-01
20	2.024E-01	1.362E-03	1.360E-03	2.712E-02	2.772E-02	1.133E-01	1.138E-01
19	1.500E-01	7.611E-04	7.602E-04	3.112E-02	3.172E-02	1.208E-01	1.215E-01
18	9.804E-02	6.444E-04	6.443E-04	3.467E-02	3.527E-02	1.268E-01	1.268E-01
17	7.950E-02	6.944E-04	6.941E-04	3.765E-02	3.823E-02	1.309E-01	1.314E-01
16	5.656E-02	7.728E-04	7.725E-04	4.125E-02	4.181E-02	1.363E-01	1.363E-01
15	4.087E-02	8.225E-04	8.215E-04	4.488E-02	4.543E-02	1.413E-01	1.411E-01
14	2.850E-02	1.012E-03	1.009E-03	4.890E-02	4.946E-02	1.463E-01	1.462E-01
13	1.931E-02	1.323E-03	1.320E-03	5.513E-02	5.560E-02	1.543E-01	1.537E-01
12	9.119E-03	1.780E-03	1.780E-03	6.245E-02	6.290E-02	1.630E-01	1.625E-01
11	5.531E-03	2.336E-03	2.326E-03	6.947E-02	6.982E-02	1.714E-01	1.706E-01
10	3.035E-03	3.399E-03	3.382E-03	8.002E-02	8.028E-02	1.821E-01	1.815E-01
9	1.234E-03	4.765E-03	4.745E-03	9.085E-02	9.885E-02	1.930E-01	1.890E-01
8	7.485E-04	6.090E-03	6.050E-03	9.960E-02	9.845E-02	2.005E-01	1.975E-01
7	4.540E-04	8.290E-03	8.137E-03	1.110E-01	1.067E-01	2.097E-01	2.003E-01
6	2.145E-04	1.193E-02	1.153E-02	1.260E-01	1.207E-01	2.203E-01	2.100E-01
5	1.013E-04	1.707E-02	1.600E-02	1.423E-01	1.293E-01	2.300E-01	2.077E-01
4	4.785E-05	2.891E-02	2.872E-02	1.690E-01	1.665E-01	2.413E-01	2.375E-01
3	1.068E-05	6.345E-02	6.291E-02	2.102E-01	2.069E-01	2.446E-01	2.405E-01
2	1.125E-06	1.055E-01	7.955E-02	2.270E-01	1.615E-01	2.223E-01	1.582E-01
1	4.140E-07	1.526E-01	2.914E-06	1.324E-01	7.556E-07	1.055E-01	5.181E-07

Grp	5"	5" w/Cd	8"	10"	12"	15"	18"
54	3.395E-02	3.420E-02	8.705E-02	1.110E-01	1.220E-01	1.215E-01	1.095E-01
53	3.233E-02	3.548E-02	9.075E-02	1.155E-01	1.260E-01	1.242E-01	1.110E-01
52	3.420E-02	3.720E-02	9.560E-02	1.210E-01	1.310E-01	1.280E-01	1.120E-01
51	3.350E-02	3.925E-02	1.017E-01	1.280E-01	1.380E-01	1.325E-01	1.150E-01
50	3.960E-02	4.180E-02	1.085E-01	1.355E-01	1.445E-01	1.360E-01	1.160E-01
49	4.285E-02	4.470E-02	1.140E-01	1.390E-01	1.450E-01	1.320E-01	1.080E-01
48	5.070E-02	5.247E-02	1.307E-01	1.563E-01	1.593E-01	1.410E-01	1.120E-01
47	5.870E-02	6.050E-02	1.450E-01	1.690E-01	1.680E-01	1.435E-01	1.100E-01
46	6.778E-02	6.948E-02	1.611E-01	1.829E-01	1.782E-01	1.468E-01	1.086E-01
45	7.952E-02	8.120E-02	1.780E-01	1.945E-01	1.825E-01	1.415E-01	9.875E-02
44	8.835E-02	9.000E-02	1.900E-01	2.020E-01	1.830E-01	1.345E-01	8.805E-02
43	9.560E-02	9.720E-02	1.950E-01	2.000E-01	1.750E-01	1.210E-01	7.420E-02
42	1.055E-01	1.070E-01	1.975E-01	1.915E-01	1.585E-01	1.006E-01	5.675E-02
41	1.185E-01	1.200E-01	2.125E-01	2.010E-01	1.620E-01	9.870E-02	5.360E-02
40	1.287E-01	1.300E-01	2.190E-01	1.997E-01	1.553E-01	8.963E-02	4.580E-02
39	1.416E-01	1.430E-01	2.319E-01	2.064E-01	1.564E-01	8.655E-02	4.231E-02
38	1.518E-01	1.530E-01	2.352E-01	2.012E-01	1.464E-01	7.619E-02	3.481E-02
37	1.625E-01	1.640E-01	2.405E-01	2.000E-01	1.410E-01	6.965E-02	3.010E-02
36	1.730E-01	1.737E-01	2.410E-01	1.930E-01	1.307E-01	6.053E-02	2.443E-02
35	1.820E-01	1.830E-01	2.405E-01	1.850E-01	1.200E-01	5.200E-02	1.950E-02
34	1.910E-01	1.917E-01	2.377E-01	1.760E-01	1.097E-01	4.443E-02	1.557E-02
33	2.022E-01	2.022E-01	2.315E-01	1.620E-01	9.530E-02	3.535E-02	1.125E-02
32	2.102E-01	2.107E-01	2.248E-01	1.497E-01	8.400E-02	2.883E-02	8.475E-03
31	2.150E-01	2.150E-01	2.195E-01	1.420E-01	7.695E-02	2.515E-02	7.040E-03
30	2.203E-01	2.207E-01	2.120E-01	1.320E-01	6.897E-02	2.127E-02	5.640E-03
29	2.257E-01	2.257E-01	2.027E-01	1.210E-01	6.027E-02	1.747E-02	4.363E-03
28	2.295E-01	2.285E-01	1.945E-01	1.120E-01	5.400E-02	1.495E-02	3.575E-03
27	2.315E-01	2.310E-01	1.885E-01	1.055E-01	4.945E-02	1.320E-02	3.080E-03
26	2.330E-01	2.320E-01	1.820E-01	9.910E-02	4.550E-02	1.180E-02	2.690E-03
25	2.340E-01	2.340E-01	1.750E-01	9.330E-02	4.190E-02	1.060E-02	2.370E-03
24	2.350E-01	2.340E-01	1.685E-01	8.780E-02	3.865E-02	9.535E-03	2.100E-03
23	2.350E-01	2.340E-01	1.597E-01	8.024E-02	3.436E-02	8.244E-03	1.790E-03
22	2.345E-01	2.335E-01	1.484E-01	7.178E-02	2.985E-02	6.966E-03	1.485E-03
21	2.325E-01	2.315E-01	1.383E-01	6.498E-02	2.643E-02	6.060E-03	1.285E-03
20	2.293E-01	2.282E-01	1.270E-01	5.793E-02	2.310E-02	5.218E-03	1.100E-03
19	2.241E-01	2.227E-01	1.144E-01	5.034E-02	1.969E-02	4.399E-03	9.261E-04
18	2.194E-01	2.176E-01	1.050E-01	4.543E-02	1.756E-02	3.907E-03	8.228E-04
17	2.158E-01	2.141E-01	9.945E-02	4.230E-02	1.626E-02	3.610E-03	7.594E-04
16	2.118E-01	2.098E-01	9.364E-02	3.934E-02	1.508E-02	3.335E-03	7.022E-04
15	2.085E-01	2.072E-01	8.899E-02	3.704E-02	1.415E-02	3.129E-03	6.585E-04
14	2.060E-01	2.044E-01	8.502E-02	3.509E-02	1.336E-02	2.952E-03	6.213E-04
13	2.030E-01	2.020E-01	8.050E-02	3.293E-02	1.250E-02	2.763E-03	5.810E-04
12	2.015E-01	1.995E-01	7.660E-02	3.110E-02	1.175E-02	2.600E-03	5.470E-04
11	2.000E-01	1.984E-01	7.375E-02	2.973E-02	1.123E-02	2.483E-03	5.224E-04
10	1.984E-01	1.964E-01	7.033E-02	2.817E-02	1.064E-02	2.344E-03	4.937E-04
9	1.965E-01	1.920E-01	6.735E-02	2.685E-02	1.010E-02	2.230E-03	4.700E-04
8	1.945E-01	1.905E-01	6.530E-02	2.595E-02	9.765E-03	2.150E-03	4.530E-04
7	1.920E-01	1.823E-01	6.273E-02	2.483E-02	9.343E-03	2.060E-03	4.340E-04
6	1.853E-01	1.787E-01	5.970E-02	2.353E-02	8.850E-03	1.953E-03	4.107E-04
5	1.837E-01	1.650E-01	5.660E-02	2.227E-02	8.370E-03	1.843E-03	3.887E-04
4	1.742E-01	1.708E-01	5.191E-02	2.035E-02	7.646E-03	1.685E-03	3.550E-04
3	1.517E-01	1.490E-01	4.348E-02	1.700E-02	6.384E-03	1.408E-03	2.966E-04
2	1.238E-01	8.725E-02	3.465E-02	1.358E-02	5.093E-03	1.125E-03	2.370E-04
1	5.385E-02	2.553E-07	1.512E-02	5.942E-03	2.243E-03	4.967E-04	1.050E-04

Appendix C: AFITBUNKI Sample Output

***** AFITBUNKI *****

OUTPUT FILE SAVED AS D20 : D20 MOD CF
 BALL DATA FILE IS DMC.C6B
 ITERATIONS BETWEEN END TESTS WAS 100
 SPECIFIED END TEST WAS 1.0%

INITIAL SPECTRUM	MAXWELL TEMP, SHAPE	CALIB. FACTOR	SMOOTH FACTOR	PER CENT ERROR	NO. OF ITERATIONS
D20-MOD CF	0.00, 0.00	1.0000	0.0000	0.1584	100

DETECTORS	MEASURED COUNTS	CALCULATED COUNTS	PERCENT DIFFERENCE
bare	4.030	4.024	-0.137
2 inch	27.190	27.279	0.327
3 inch	53.660	53.650	-0.019
5 inch	65.690	65.583	-0.163
8 inch	36.300	36.258	-0.117
10 inch	21.390	21.393	0.014
12 inch	12.650	12.662	0.096

NEUTRON SPECTRUM DATA

TOTAL FLUENCE (F) =	3.524E+02	neut/cm ²
AVG ENERGY (INCL THERM, E-> 1E-11 MeV) =	5.583E-01	MeV
AVG ENERGY (LESS THERM, E-> 0.414 eV) =	5.607E-01	MeV

TRADITIONAL DOSE/DOSE EQUIVALENT QUANTITIES

DOSE (D) (ICRP 21) =	4.627E+03	pGy
DOSE EQUIVALENT (H) (ICRP 21) =	3.263E+04	pSv
AVERAGE QUALITY FACTOR (Q) (ICRP 21) =	7.052	pSv/pGy
DOSE EQUIVALENT/FLUENCE (H/F) =	9.259E+01	pSv-cm ² /n

TRADITIONAL EFFECTIVE DOSE EQUIVALENT QUANTITIES

EFFECTIVE DOSE EQUIVALENT (HE) (ICRP 26/51),		
(PRE-1985 QUALITY FACTORS) AP EXPOSURE =	1.841E+04	pSv
ROT EXPOSURE =	1.059E+04	pSv

VOLUMETRIC DOSE QUANTITIES

EFFECTIVE DOSE (E) (ICRP 60) AP EXPSR =	3.424E+04	pSv
ISOTROPIC EXPOSURE =	1.690E+04	pSv

PRE-1985 ENVIRONMENTAL MONITORING QUANTITIES

AMBIENT DOSE (D*[10]) (ICRP 26) =	3.962E+03	pGy
AMBIENT DOSE EQUIV (H*[10]) (ICRP 26) =	3.362E+04	pSv
EFF AMB QUALITY FACTOR (Q*) (ICRP 26) =	8.487	pSv/pGy
AMB DOSE EQIV/FLUENCE (H*/F) (ICRP 26) =	9.541E+01	pSv-cm ² /n

POST-1985 ENVIRONMENTAL MONITORING QUANTITIES

AMB DOSE EQUIVALENT (H*[10]) (ICRP 60) =	4.503E+04	pSv
EFF AMB QUALITY FACTOR (Q*) (ICRP 60) =	11.365	pSv/pGy
AMB DOSE EQIV/FLUENCE (H*/F) (ICRP 60) =	1.278E+02	pSv-cm ² /n

***** AFITBUNKI *****
 OUTPUT FILE SAVED AS D2O : D2O MOD CF

BIN NO.	ENERGY MAX (MeV)	FLUENCE n/cm2	FLUENCE n/cm2/leth	DOSE pGy	HsubE[AP] % of Tot	H*(10) [60] % of Tot
1	4.140E-07	1.526E+00	3.305E-01	8.194E+00	3.60E-02	4.25E-02
2	1.125E-06	1.197E+00	2.758E+00	7.496E+00	3.10E-02	3.38E-02
3	1.068E-05	3.218E+01	3.292E+01	1.986E+02	8.09E-01	8.54E-01
4	4.785E-05	2.424E+01	3.722E+01	1.446E+02	5.72E-01	5.77E-01
5	1.013E-04	1.330E+01	4.084E+01	7.767E+01	3.03E-01	2.90E-01
6	2.145E-04	1.442E+01	4.427E+01	8.185E+01	3.18E-01	2.95E-01
7	4.540E-04	1.543E+01	4.740E+01	8.424E+01	3.29E-01	2.97E-01
8	7.485E-04	1.103E+01	5.081E+01	5.826E+01	2.31E-01	2.03E-01
9	1.234E-03	1.189E+01	5.478E+01	6.120E+01	2.48E-01	2.14E-01
10	3.035E-03	2.262E+01	5.789E+01	1.126E+02	4.85E-01	4.09E-01
11	5.531E-03	1.610E+01	6.178E+01	7.837E+01	3.62E-01	3.11E-01
12	9.119E-03	1.404E+01	6.467E+01	6.874E+01	3.32E-01	3.12E-01
13	1.931E-02	2.226E+01	6.830E+01	1.146E+02	6.11E-01	6.90E-01
14	2.850E-02	1.181E+01	6.988E+01	6.559E+01	4.15E-01	5.64E-01
15	4.087E-02	1.073E+01	6.851E+01	6.378E+01	4.82E-01	7.55E-01
16	5.656E-02	9.572E+00	6.783E+01	6.129E+01	5.52E-01	1.01E+00
17	7.950E-02	9.925E+00	6.712E+01	6.912E+01	7.48E-01	1.57E+00
18	9.804E-02	5.902E+00	6.483E+01	4.443E+01	5.6E-01	1.31E+00
19	1.500E-01	1.144E+01	6.196E+01	9.701E+01	1.48E+00	3.59E+00
20	2.024E-01	7.626E+00	5.861E+01	7.571E+01	1.40E+00	3.43E+00
21	2.472E-01	4.743E+00	5.462E+01	5.322E+01	1.10E+00	2.62E+00
22	2.985E-01	4.156E+00	5.075E+01	5.174E+01	1.15E+00	2.63E+00
23	3.688E-01	4.099E+00	4.463E+01	5.718E+01	1.36E+00	2.92E+00
24	4.076E-01	1.587E+00	3.653E+01	2.430E+01	6.04E-01	1.23E+00
25	4.505E-01	1.170E+00	2.692E+01	1.908E+01	4.85E-01	9.45E-01
26	4.979E-01	1.118E+00	2.573E+01	1.945E+01	5.05E-01	9.40E-01
27	5.502E-01	1.576E+00	3.632E+01	2.932E+01	7.72E-01	1.37E+00
28	6.081E-01	1.994E+00	4.589E+01	3.982E+01	1.05E+00	1.80E+00
29	7.065E-01	3.188E+00	4.895E+01	6.992E+01	1.85E+00	2.98E+00
30	8.209E-01	3.282E+00	5.035E+01	8.106E+01	2.12E+00	3.18E+00
31	9.072E-01	2.040E+00	4.699E+01	5.584E+01	1.44E+00	2.03E+00
32	1.003E+00	1.610E+00	3.693E+01	4.779E+01	1.21E+00	1.63E+00
33	1.225E+00	3.084E+00	3.551E+01	1.010E+02	2.56E+00	3.19E+00
34	1.423E+00	3.019E+00	4.640E+01	1.087E+02	2.79E+00	3.19E+00
35	1.572E+00	2.447E+00	5.659E+01	9.349E+01	2.43E+00	2.61E+00
36	1.827E+00	4.217E+00	6.459E+01	1.695E+02	4.49E+00	4.54E+00
37	2.019E+00	3.002E+00	6.918E+01	1.63E+02	3.42E+00	3.25E+00
38	2.307E+00	4.463E+00	7.707E+01	1.945E+02	5.40E+00	4.85E+00
39	2.592E+00	4.840E+00	9.568E+01	2.180E+02	6.26E+00	5.28E+00
40	3.012E+00	6.442E+00	9.877E+01	2.996E+02	8.91E+00	7.04E+00
41	3.329E+00	3.606E+00	8.298E+01	1.725E+02	5.31E+00	3.94E+00
42	4.066E+00	5.794E+00	6.671E+01	2.859E+02	9.11E+00	6.32E+00
43	4.493E+00	2.489E+00	5.739E+01	1.266E+02	4.17E+00	2.71E+00
44	4.966E+00	2.456E+00	5.649E+01	1.273E+02	4.27E+00	2.67E+00
45	6.065E+00	4.148E+00	4.777E+01	2.213E+02	7.59E+00	4.48E+00
46	7.047E+00	2.135E+00	3.276E+01	1.179E+02	4.14E+00	2.29E+00
47	7.788E+00	9.408E-01	2.167E+01	5.322E+01	1.90E+00	1.00E+00
48	9.048E+00	8.515E-01	1.307E+01	4.937E+01	1.78E+00	9.04E-01
49	1.000E+01	3.440E-01	7.918E+00	2.046E+01	7.52E-01	3.64E-01
50	1.105E+01	1.884E-01	4.345E+00	1.145E+01	4.29E-01	1.99E-01
51	1.221E+01	7.649E-02	1.764E+00	4.756E+00	1.83E-01	8.16E-02
52	1.284E+01	1.958E-02	8.961E-01	1.239E+00	4.85E-02	2.13E-02
53	1.419E+01	2.909E-02	6.700E-01	1.875E+00	7.45E-02	3.31E-02
54	1.492E+01	8.548E-03	3.923E-01	5.612E-01	2.26E-02	1.03E-02

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Vita

Captain Sean C. Miller was born on 8 February 1962 in Lincoln, Nebraska. He graduated from Barnstable High School in Hyannis, Massachusetts in 1979. In 1987 he graduated from Wright State University, Dayton, Ohio, with a Bachelor of Science in Physics and received his commission through ROTC. In the spring of 1988 he went on extended active duty and ended up in the Space Surveillance and Tracking System (currently Brilliant Eyes) SPO of Air Force SDIO Programs at Los Angeles AFB, California. His first assignment there involved him in an OSD-directed study to baseline the current ground-based Space Surveillance Network to assess the network's capability to support an advanced ASAT weapon system. He was then appointed Test Manager for the Saipan Radar Program where he was responsible for the successful completion of DT&E and IOT&E before the system was turned over to AFSPACECOM. In July 1990, he was selected to serve as the Brilliant Eyes Program Director's Executive Officer until he entered the School of Engineering, Air Force Institute of Technology, in August 1991.

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